Y3.N88:25/2497/v.2

NUREG/CR-2497
Volume 2
ORNL/NSIC-182/V2

COMPLETED
1079

ORIGINAL

Precursors to Potential Severe Core Damage Accidents: 1969-1979
A Status Report

Appendix B

J. W. Minarick* C. A. Kukielka*

*Science Applications, Inc.
Oak Ridge National Laboratory

Prepared for
U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research

I

NOTICE

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, or any of their employees, makes any warranty, expressed or implied, or assumes any legal liability or responsibility for any third party's use, or the results of such use, of any information, apparatus, product or process disclosed in this report, or represents that its use by such third party would not infringe privately owned rights.

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 1717 H Street, N.W.
   Washington, DC 20555
2. The NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission,
   Washington, DC 20555
3. The National Technical Information Service, Springfield, VA 22161

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC Office of Inspection and Enforcement bulletins, circulars, information notices, inspection and investigation notices; Licensee Event Reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the NRC/GPO Sales Program: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, and NRC booklets and brochures. Also available are Regulatory Guides, NRC regulations in the *Code of Federal Regulations*, and *Nuclear Regulatory Commission Issuances*.

Documents available from the National Technical Information Service include NUREG series reports and technical reports prepared by other federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal and periodical articles, and transactions. *Federal Register* notices, federal and state legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free upon written request to the Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland, and are available there for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018.

GPO Printed copy price: $13.00

II

NUREG/CR-2497
Volume 2
ORNL/NSIC-182/V2
Dist. Category AN,
RG, XA, 1S

Contract No. W-7405-eng-26

Nuclear Safety Information Center

PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS: 1969-1979
A STATUS REPORT

Appendix B

J. W. Minarick   C. A. Kukielka
Science Applications, Inc.

Manuscript Completed - April 1982
Date Published - June 1982

Prepared for the
U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, D. C. 20555

NRC FIN No. B1583

Prepared by the
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37830
operated by
UNION CARBIDE CORPORATION
for the
DEPARTMENT OF ENERGY

III

NOTE

This document is bound in two volumes: Vol. 1 contains the main report and Appendixes A, C, D, and E; Vol. 2 contains Appendix B.

Appendix B

PRECURSOR SUMMARY SHEETS AND EVENT TREES

One hundred and sixty-nine PWR and BWR events for the years 1969 through 1979 were selected as precursors to potential severe core damage, based on the selection methods and criteria described in Vol. 1 of this report. These precursors are documented herein. For each precursor, four items are included.

1. A precursor description and data sheet lists the report date, briefly describes the LER event, identifies what corrective action was taken after the event, describes the purpose of the failed system or component, and lists the unavailability of equivalent systems and components as determined in the *Reactor Safety Study*.*

2. An actual-occurrence event tree describes the major functions associated with the reported event which maintained the plant in a safe condition. Failed items on these trees are identified by hatch marks. The actual sequence of the event is indicated by arrows on the appropriate tree branches.

3. A sequence-of-interest event tree describes safety-related functions required to prevent severe core damage for a postulated initiating event. The choice of postulated events is discussed in Vol. 1 of this report. Four degraded or failed-function states are indicated on the sequence-of-interest tree:

- degraded (D) — when a safety-related function is degraded;
- consequently degraded (CD) — when a safety-related function is degraded as a consequence of a degraded or failed associated function, for example, when an auxiliary feedwater function is degraded during a loss of offsite power because of a diesel generator failure;
- failed (F) — when a safety-related function fails;
- consequently failed (CF) — when a safety-related function fails as a consequence of a failed or degraded associated function, for example, when an auxiliary feedwater function fails because of the failure of both plant emergency diesels during a loss of offsite power (if all auxiliary feedwater pumps are motor driven).

In addition to the four states discussed above, arrows are used in the sequence-of-interest event trees to indicate an inherent unavailability of a particular function at a plant, for example, turbine runback capability following a load rejection from 100% power (which does not exist on all PWRs).

4. A categorization of accident sequence precursors sheet lists characteristics identified for each precursor. These characteristics have been used in trends analyses. In this listing, the entry for DURATION

---

*U.S. Nuclear Regulatory Commission, *Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants*, WASH-1400 (NUREG-75/014) (October 1975).

includes an (a) if an assumed value (based on testing interval) was used because an event-specific value was unavailable.

The 169 precursor descriptions are arranged in numerical order of the associated NSIC accession number. Four indexes (Table B.1) are provided as an aid in using this appendix. These indexes are ordered by (1) NSIC accession number (ACCESS), (2) date of event (E DATE), (3) sequence of interest (SEQ), and (4) name of reactor (PLANT).

Table B.1. Indexes for potential precursor descriptions

TABLE COLUMN HEADING ABBREVIATIONS:
ACCESS: 6 DIGIT NSIC ACCESSION NUMBER
E DATE: EVENT DATE
SEQ: SEQUENCE OF INTEREST FOR THE EVENT
ECIT - EXCESSIVE COOLANT INVENTORY
EQUK - EARTHQUAKE
INAA - INADVERTANT ADS ACTUATION
LOFW - LOSS OF FEEDWATER
LOOP - LOSS OF OFFSITE POWER
LOCA - LOSS OF COOLANT ACCIDENT
LRTR - LOCKED ROTOR ACCIDENT
MSLB - MAIN STEAM LINE BREAK
RCPT - REACTOR COOLANT PUMP TRIP
SGTR - STEAM GENERATOR TUBE RUPTURE
ACTUAL OCCURRENCE: DESCRIPTION OF EVENT
PLANT NAME: NAME OF PLANT AND UNIT NUMBER
PG: PAGE NUMBER

FILE SORTED BY ACCESSION NUMBER

ACCESS	E DATE	SEQ	ACTUAL OCCURRENCE	PLANT	PG
032429	690313	MSLB	LOOP ISOLAT. VALVES FAILED TO SHUT	IND.POINT1	12
036147	690715	LOOP	REACTOR TRIP WITH LOSS OF OFFSITE POWER	HAD.NECK	16
039024	720125	LOOP	LOSS OF OFFSITE POWER	BG ROCK PT	20
039380	711208	ECIT	SAFETY VALVE OPERATION AFTER LOFW	DRESDEN 3	25
044751	700427	LOFW	3 OF 4 STEAM DRUM SCRAM SENSORS FAIL	DRESDEN 1	29
047814	700505	ECIT	DEPRESSURIZATION INCIDENT	DRESDEN 2	33
059484	710112	LOCA	SUMP ISO. VALVES CLOSED	PT.BEACH 1	37
060227	701204	MSLB	FAILURE OF SEVERAL MSIV'S TO CLOSE	DRESDEN 2	41
061043	710120	LOOP	LOSS OF OFFSITE POWER	LACROSSE	45
061434	700717	LOOP	LOSS OF OFFSITE POWER	HUMBDLTBAY	49
061565	710902	LOOP	LOOP AND FAILURE OF A DIESEL GENERATOR TO LOAD	PALISADES	54
063129	710324	LOOP	LOSS OF OFFSITE POWER	LACROSSE	58
063144	710308	LOOP	FAILURE OF BOTH DIESEL GENER. DURING TESTING	ROBINSON 2	63
064600	701226	LOOP	SHUTDOWN OF BUCHANAN STATION	IND.POINT1	67
065757	710205	LOOP	LOSS OF OFFSITE POWER	PT.BEACH 1	71
065757	710416	LOCA	FAILURE OF CONTAINMENT SUMP ISO VALVES	PT.BEACH 1	75
065969	710908	LOCA	OPEN ELECTROMATIC RELIEF VALVE	PALISADES	79
066996	711010	LOCA	TRANSIENT AND BLOWDOWN	MILLSTONE1	84
071694	720517	LOOP	LOSS OF OFFSITE POWER	PALISADES	89
073655	720720	LOOP	LOSS OF OFFSITE POWER	IND.POINT1	93
074242	711231	ECIT	HIGH COOLANT LEVEL	NINEMIPT1	97
075074	720817	LOOP	LOSS OF LOAD	LACROSSE	101
077916	721229	LOCA	MALFUNCTION OF SEVERAL VALVES	OYSTER CRK	105
078418	730202	LOCA	VALVE FAILURE TO OPEN DURING TESTING	MA.YANKEE	110
079565	721201	LOOP	LOSS OF NORMAL STATION POWER	VT.YANKEE	114
080138	720610	LOOP	FLOODING OF TURBINE BUILDING	QUAD-CTES1	118
081523	730618	LOFW	FAILURE OF AFW PUMPS TO AUTO-START	TKY.POINT4	123
082198	730704	LOOP	REACTOR/TURBINE TRIP DURING TEST	TKY.POINT3	127
083833	730908	LOOP	LOSS OF OFFSITE POWER	OYSTER CRK	131
085370	731021	LOOP	LOOP, EXCESSIVE RCS COOLDOWN, SFTY INJ.& INS BUS	GINNA	137
085566	731119	LOOP	LOSS OF A.C. POWER CAUSE HPCI/RCIC TO FAIL	BRN.FERRY1	143
085738	731119	LOFW	RCIC/HPCI FAILS DURING TESTING	BRN.FERRY1	147
086990	731121	LOOP	WORKER BUMPS RELAY CAUSING LOOP	NINEMIPT1	151
088451	740119	LOOP	LOSS OF OFFSITE POWER	HAD.NECK	155
089153	740221	MSLB	OPEN RWST ISOLATION VALVES	PRAIRIEIS1	159
089205	731130	LRTR	FAILED RCP SHAFT AT 92% POWER	SURRY 1	163
089252	740217	LOOP	FAILURE OF DIESEL TO START/LOAD;2 AFW PMPS OUT	ZION 2	167
090421	740407	LOFW	INOPERABLE AFW PUMPS DURING PLANT SHUTDOWN	PT.BEACH 1	171
091676	740508	LOFW	FAILURE OF 3 AUX FWDTR PMPS TO START AT TEST	TKY.POINT3	175
093553	741208	LOOP	TWO DIESEL GENERATORS MALFUNCTION	HATCH 1	179
093702	740425	LOOP	REACTOR TRIP WITH LOSS OF OFFSITE POWER	TKY.POINT4	183
095040	740823	LOFW	HPCI INJECTION VALVE AND RCIC FAIL	BRN.FERRY2	188
097107	741022	MSLB	BIT AUTO INJECTION FAILURE	ZION 2	192

[redacted]

Table B.1 (continued)

097578	741017	LOOP	LOSS OF OFFSITE POWER DURING TESTING	PALISADES	196
097779	741124	RCPT	FAILURE OF RPS TO SEE RCP TRIPS	ARKANCAS 1	200
100415	750225	LOFW	ALL DRYWELL HIGH PRESSURE SWITCHES ISOLATED	FITZPATRIK	205
101078	750318	LOOP	DIESEL GENERATOR FIRE	FITZPATRIK	209
101444	750322	LOFW	CABLE TRAY FIRE CAUSED EXTENSIVE DAMAGE	BRN.FERRY1	213
102146	750320	LOOP	LOSS OF OFFSITE POWER	BRUNSWICK2	220
103002	750429	LOFW	MULTIPLE VALVE FAILURES & RCIC INOPERABLE	BRUNSWICK2	224
103077	750501	LOCA	RCP SEAL FAILURE	ROBINSON 2	228
103111	750419	LOOP	CONDENSER EMERG. DISCHARGE VLVS FAIL TO OPEN	OCONEE 3	233
103207	750525	LOOP	BREAKER & RELAY FAILURE DURING TEST	TKY.POINT4	237
103892	750613	LOCA	STUCK OPEN PORV	OCONEE 3	241
104087	750619	LOCA	CLOSED SI TANK ISOLATION VALVES	MA.YANKEE	245
104379	750627	EQUK	EARTHQUAKE OCCURS	HUMBDLTBAY	249
104817	750729	INAA	ALL ADS VALVES OPEN INADVERTENTLY	HATCH 1	253
105540	750817	LOCA	LOW FLOW FEEDWATER LINE SEVERED AT 6X4 REDUCER	QUAD-CTES2	257
106333	750909	LOFW	HPCI FOUND INOPERABLE	QUAD-CTES2	261
106616	750913	LOOP	LOSS OF OFFSITE PWR & RELF.VALV STKS OPN	PILGRIM 1	265
108078	751105	LOFW	INOPERABLE AFW PUMPS DURING PLANT STARTUP	KEWAUNEE	269
108869	751220	LOOP	TWO DIESELS FAIL TO START	HATCH 1	273
109295	751219	LOCA	FROZEN BWST LEVEL INPULSE LINES WHILE OPRTING	OCONEE 1	277
112309	760308	LOOP	GAS TURBINE BECOMES INOPERABLE	MILLSTONE1	281
112310	760315	LOOP	GAS TURBINE FAILS TO ACHIEVE RATED SPEED	MILLSTONE1	285
115870	760706	LOCA	IMPROPER INSULATION	VT.YANKEE	289
115875	760626	LOOP	LOSS OF OFFSITE POWER DURING REFUELING	HAD.NECK	293
116212	760720	LOOP	APPARENT LOOP & FAILURE OF SAFETY RELATED COMP	MILLSTONE2	297
116780	760810	LOOP	GAS TURBINE FAILS DURING PLANT TRIP	MILLSTONE1	302
117944	760810	LOOP	REACTOR TRIP WITH LOOP & DIESEL GENS.RUNNING	MILLSTONE2	306
120207	761028	MSLB	FAILURE OF RWST-CHARGING PUMP SUCTION VALVES	TROJAN	310
120293	761111	LOOP	PLANT SERVICE WATER STRAINERS PLUGGED	HATCH 1	314
120439	761101	LOFW	TWO ELECTROMATIC RV FAILURES	QUAD-CTES1	318
120443	761101	LOFW	TWO ELECTROMATIC RV FAILURES	QUAD-CTES2	323
120995	761111	LOCA	VALVES FAIL TO OPEN FOR DECAY HEAT SYSTEM	RANCHOSECO	326
121533	770104	LOOP	TWO DIESEL GENERATORS TRIP	BRUNSWICK1	330
122366	770110	MSLB	SENSING LINES FROZEN	ARKANSAS 1	335
123118	770218	LOFW	BOTH AFW PUMPS UNAVAILABLE	RANCHOSECO	339
123150	770303	MSLB	50% OPEN ATMOSPHERIC DUMP VALVES	CRYSTALRV3	343
124201	770322	LOOP	DIESEL GENERATOR FIELD FAILS TO FLASH	DRESDEN 3	347
124222	770408	LOFW	SIX MAIN STEAM RELIEF VALVES FAIL TO LIFT	D.ARNOLD	351
124920	770416	MSLB	STEAM GENERATOR OVERFILL	CRYSTALRV3	355
125106	770506	LOCA	FAILURE TO CLEAR OVERPRESSURE DURING HEATUP	SALEM 1	359
125563	770516	LOOP	REACTOR TRIP WITH LOSS OF OFFSITE POWER	ST.LUCIE 1	363
127010	770714	LOOP	FAILURE OF TWO DIESEL GENERATORS	DRESDEN 3	367
127384	770712	LOCA	DUMMY SIGNALS INSTALLED ON INSTRUMENTENTATION	ZION 2	371
128569	770715	LOCA	SAFETY RELIEF VALVE FAILS TO RESET	BRUNSWICK2	376
128906	770831	LOFW	LOSS OF NO-BREAK-POWER AND FEEDWATER CONTROL	COOPER	381
128935	770822	LOOP	MOMENTARY LOSS OF OFFSITE POWER	FTCALHOUN1	386
130111	770901	LOOP	REACTOR TRIP & LOSS OF OFFSITE POWER	COOK 1	391
130119	770924	LOOP	REACTOR TRIP DUE TO LOSS OF OFFSITE POWER	PALISADES	395
130788	770924	LOCA	STUCK OPEN PORV	DVS-BESSE1	399
130889	771011	LOCA	FLOODING OF VALVE PIT DURING 0 PWR PHYSIC TST	SURRY 2	403
131225	770828	LOOP	TWO DIESEL GENERATORS FAIL TO START	PEACHBTOM2	407
132155	771129	LOOP	DIESEL GENERATOR TRIPS DURING TESTING	DRESDEN 3	411
132927	771129	LOOP	LOSS OF OFFSITE POWER	DVS-BESSE1	415
132942	771209	LOOP	BOTH DIESEL GENERATORS INOPERABLE	PRAIRIEIS1	419
132943	771125	LOOP	REACTOR TRIP WITH LOSS OF OFFSITE POWER	PALISADES	424
132947	771202	LOOP	DIESEL GENERATOR FAILS TO START	DRESDEN 2	428
132958	771211	LOOP	REACTOR TRIP WITH LOSS OF OFFSITE POWER	PALISADES	432
132979	770515	LOOP	FUEL SUPPLY VALVES FOUND SHUT DURING INSPECT.	MILLSTONE2	436
133004	770613	LOOP	3/4 DIESELS INOPERABLE	PEACHBTOM3	440
133706	771211	LOFW	AUX. FEEDWATER PUMPS INOPERABLE DURING TEST	DVS-BESSE1	445
135006	770916	LOOP	LOSS OF OFFSITE POWER	FARLEY 1	449
135077	771026	MSLB	THERMO WELL COUPLING FAILED AFTER SAFETY INJ.	ROBINSON 2	454
135079	771123	LOCA	SI PUMP BREAKERS INCORRECTLY RACKED OUT	ROBINSON 2	458
135932	780206	LOOP	LOSS OF OFFSITE POWER	PILGRIM 1	462
137305	780325	LOFW	LOW-LOW WATER LEVEL IN ONE STEAM GENERATOR -	FARLEY 1	466
137335	780404	LOFW	IC INADVERNTLY RENDERED INOPERABLE	DRESDEN 2	470
137384	780318	LOCA	RELAY FAILURE	TMI 1	474
137543	780413	LOOP	LOSS OF OFFSITE POWER WHILE SHUTDOWN	CALCLIFFS1	478
137918	780423	MSLB	MULTIPLE STUCK-OPEN RELIEF VALVES	TMI 2	482
138830	780320	LOFW	FAILURE OF NNI & STEAM GENERATOR DRYOUT	RANCHOSECO	486
139565	780514	LOOP	LOSS OF OFFSITE POWER DURING REFUELING	ST.LUCIE 1	491
139573	780615	LOOP	UNAVAILABILITY OF BOTH DIESEL GENERATORS	COOK 2	495

Table B.1 (continued)

140140	780806	LOOP	LOSS OF OFFSITE POWER	PILGRIM 1	499
140162	780824	LOOP	DIESEL GENERATOR FAILS TO START	DRESDEN 2	504
140335	780728	LOOP	LOOP AND DIESEL GENERATOR FAILURE	BVRVALLEY1	508
140691	780905	LOOP	BOTH DIESEL GENERATORS OUT OF SERVICE	BVRVALLEY1	512
140693	780912	LOOP	BOTH DIESEL GENERATORS OUT OF SERVICE	BVRVALLEY1	516
141097	780329	LOCA	STUCK OPEN PORV	TMI 2	520
141455	781004	LOOP	BOTH DIESEL GENERATORS OUT OF SERVICE	RANCHOSECO	524
141523	781017	LOCA	INOPERABLE LPSI PUMPS	CALCLIFFS2	528
141706	781024	LOOP	LOSS OF 1 DG WHILE 2ND OUT OF SERVICE	RANCHOSECO	532
142329	781126	MSLB	HEAT TRACING CIRCUIT FAILURE	IND.POINT2	536
142462	781127	LOCA	LOSS OF VITAL INST. BUS-REACTOR TRIP	SALEM 1	540
143219	770713	LOOP	LOSS OF OFFSITE POWER	IND.POINT3	545
143255	770730	LOOP	DIESEL GENERATORS FAILED TO START AT TEST	SALEM 1	549
143296	771020	LOCA	INADVERTENT VALVE CLOSING DURING TEST	MA.YANKEE	553
143533	771010	LOOP	DIESEL GENERATOR LOAD CYCLES	BRUNSWICK1	557
144187	771210	LOOP	DIESEL GENERATOR FOUND INOPERABLE	MILLSTONE1	561
145209	781220	LOCA	TWO STUCK OPEN PROV'S DURING COLD SHUTDOWN	FTCALHOUN1	565
146540	790110	LOOP	ONE DG FAILED WHILE 2ND OUT DUE TO MAINTENANCE	KEWAUNEE	569
146744	790112	LOFW	LOSS OF VITAL BUS WHILE AT POWER	DVS-BESSE1	573
147223	790106	MSLB	TURBINE TRIP AND STUCK FW VALVE	CRYSTALRV3	578
147383	790223	LOOP	TWO DIESEL GENERATORS NOT AVAILABLE (TWICE)	COOK 1	582
147399	790221	LOOP	BOTH DIESEL GENERATORS UNAVAILABLE	KEWAUNEE	587
147400	790219	MSLB	MAIN STEAM STOP VALVES FAILED TO CLOSE AT TEST	PT.BEACH 2	591
148764	790118	MSLB	STUCK OPEN STEAM DUMP VALVES	BVRVALLEY1	595
148892	790222	LOOP	UNAVAILABILITY OF TWO DIESEL GENERATORS	FARLEY 1	599
149230	790321	LOOP	BOTH EMERGENCY SERVICE WATER SYSTEM OUT	D.ARNOLD	603
149250	790414	MSLB	MSIVS FAILED TO CLOSE DURING TEST	TROJAN	607
149450	790502	LOFW	LOSS OF FEEDWATER FLOW	OYSTER CRK	611
149961	790603	LOFW	HPCI FAILS TO START GIVEN LOFW	HATCH 1	616
149962	790505	LOOP	DIESEL GENERATOR UNABLE TO CARRY LOAD	HATCH 1	620
150051	790612	LOOP	FAILURE OF TWO DIESEL GENERATORS	DRESDEN 2	624
150271	790603	LOFW	BOTH RHR SERVICE WATER PMPS NOT OPERABLE	BRUNSWICK2	628
150499	790628	LOFW	RCIC AND HPCI ISOLATE FOLLOWING REACTOR SCRAM	HATCH 2	632
150742	790622	LOFW	MAIN STACK SAMPLING SYSTEM LOST	PEACHBTOM3	636
150882	790606	LOOP	BOTH DG'S TRIPPED DURING TEST	CRYSTALRV3	640
150943	790727	LOOP	LOSS OF OFFSITE POWER	PILGRIM 1	644
151433	790911	LOCA	INCORRECT RHR SERVICE PUMP BREAKER RACKED OUT	BRUNSWICK2	649
151635	790828	LOOP	LOSS OF OFFSITE POWER	PILGRIM 1	653
152183	790813	LOCA	OPEN PWR OPERATED RELIEF VALVE + ISO.VALVE	HAD.NECK	658
152187	790903	LOOP	SWITCHYARD LOCKOUT DUE TO CABLE DROP AT STORM	ST.LUCIE 1	662
152563	791002	SGTR	STEAM GENERATOR TUBE BREAK	PRAIRIEIS1	666
152638	791017	LOOP	UNAVAIL OF 1 AFW PUMP AND 1 DIESEL GENERATOR	COOK 2	671
152835	790731	LOOP	OVR VOLTG PROT TRIP FOUND WHILE TRBLE SHTG	CALCLIFFS1	675
152951	791015	LOOP	REACTOR TRIP WITH LOSS OF OFFSITE POWER	DVS-BESSE1	679
153003	791024	LOCA	FAILURE TO PROPERLY POSITION VALVES AFTER TEST	OCONEE 2	684
153164	790328	LOFW	LOSS OF FEEDWATER & OPEN PORV	TMI 2	688
153167	790926	LOOP	LOSS OF STANDBY POWER SOURCE DURING PWR OPRT	OCONEE 1	693
153167	790926	LOOP	LOSS OF STANDBY POWER SOURCE DURING PWR OPRT	OCONEE 2	693
153333	791115	LOFW	AFW UNAVAIL DUE TO MAINTENANCE AND INSPECTION	COOK 1	697
153338	791101	LOOP	INCORRECT DIESEL GENERATOR MAINT. WHILE REFUEL	COOK 2	701
153686	791127	LOOP	DG AND CHARGING PMP SUCT VALVE UNAVAILABLE	BVRVALLEY1	705
153810	791120	LOFW	RCIC TURBINE TRIP WITH HPCI UNAVAILABLE	BRUNSWICK1	709
154286	791128	LOCA	ONE PMP FAILED + 1 DEGRADED IN CLD SHTDN	SALEM 1	713
154639	791209	LOOP	BOTH DIESEL GENERATORS INOPERABLE	DVS-BESSE1	717

FILE SORTED BY EVENT DATE AND ACCESSION NUMBER

ACCESS	E DATE	SEQ	ACTUAL OCCURRENCE	PLANT	PG
032429	690313	MSLB	LOOP ISOLAT. VALVES FAILED TO SHUT	IND.POINT1	12
036147	690715	LOOP	REACTOR TRIP WITH LOSS OF OFFSITE POWER	HAD.NECK	16
044751	700427	LOFW	3 OF 4 STEAM DRUM SCRAM SENSORS FAIL	DRESDEN 1	29
047814	700505	ECIT	DEPRESSURIZATION INCIDENT	DRESDEN 2	33
061434	700717	LOOP	LOSS OF OFFSITE POWER	HUMBDLTBAY	49
060227	701204	MSLB	FAILURE OF SEVERAL MSIV'S TO CLOSE	DRESDEN 2	41
064600	701226	LOOP	SHUTDOWN OF BUCHANAN STATION	IND.POINT1	67
059484	710112	LOCA	SUMP ISO. VALVES CLOSED	PT.BEACH 1	37
061043	710120	LOOP	LOSS OF OFFSITE POWER	LACROSSE	45
065757	710205	LOOP	LOSS OF OFFSITE POWER	PT.BEACH 1	71
063144	710308	LOOP	FAILURE OF BOTH DIESEL GENER. DURING TESTING	ROBINSON 2	63
063129	710324	LOOP	LOSS OF OFFSITE POWER	LACROSSE	58
065757	710416	LOCA	FAILURE OF CONTAINMENT SUMP ISO VALVES	PT.BEACH 1	75
061565	710902	LOOP	LOOP AND FAILURE OF A DIESEL GENERATOR TO LOAD	PALISADES	54
065969	710908	LOCA	OPEN ELECTROMATIC RELIEF VALVE	PALISADES	79

Table B.1 (continued)

066996	711010	LOCA	TRANSIENT AND BLOWDOWN	MILLSTONE1	84
039380	711208	ECIT	SAFETY VALVE OPERATION AFTER LOFW	DRESDEN 3	25
074242	711231	ECIT	HIGH COOLANT LEVEL	NINEMIPT1	97
039024	720125	LOOP	LOSS OF OFFSITE POWER	BG ROCK PT	20
071694	720517	LOOP	LOSS OF OFFSITE POWER	PALISADES	89
080138	720610	LOOP	FLOODING OF TURBINE BUILDING	QUAD-CTES1	118
073655	720720	LOOP	LOSS OF OFFSITE POWER	IND.POINT1	93
075074	720817	LOOP	LOSS OF LOAD	LACROSSE	101
079565	721201	LOOP	LOSS OF NORMAL STATION POWER	VT.YANKEE	114
077916	721229	LOCA	MALFUNCTION OF SEVERAL VALVES	OYSTER CRK	105
078418	730202	LOCA	VALVE FAILURE TO OPEN DURING TESTING	MA.YANKEE	110
081523	730618	LOFW	FAILURE OF AFW PUMPS TO AUTO-START	TKY.POINT4	123
082198	730704	LOOP	REACTOR/TURBINE TRIP DURING TEST	TKY.POINT3	127
083833	730908	LOOP	LOSS OF OFFSITE POWER	OYSTER CRK	131
085370	731021	LOOP	LOOP,EXCESSIVE RCS COOLDOWN,SFTY INJ.&INS BUS	GINNA	137
085566	731119	LOOP	LOSS OF A.C. POWER CAUSE HPCI/RCIC TO FAIL	BRN.FERRY1	143
085738	731119	LOFW	RCIC/HPCI FAILS DURING TESTING	BRN.FERRY1	147
086990	731121	LOOP	WORKER BUMPS RELAY CAUSING LOOP	NINEMIPT1	151
089205	731130	LRTR	FAILED RCP SHAFT AT 92% POWER	SURRY 1	163
088451	740119	LOOP	LOSS OF OFFSITE POWER	HAD.NECK	155
089252	740217	LOOP	FAILURE OF DIESEL TO START/LOAD;2 AFW PMPS OUT	ZION 2	167
089153	740221	MSLB	OPEN RWST ISOLATION VALVES	PRAIRIEIS1	159
090421	740407	LOFW	INOPERABLE AFW PUMPS DURING PLANT SHUTDOWN	PT.BEACH 1	171
093702	740425	LOOP	REACTOR TRIP WITH LOSS OF OFFSITE POWER	TKY.POINT4	183
091676	740508	LOFW	FAILURE OF 3 AUX FWDTR PMPS TO START AT TEST	TKY.POINT3	175
095040	740823	LOFW	HPCI INJECTION VALVE AND RCIC FAIL	BRN.FERRY2	188
097578	741017	LOOP	LOSS OF OFFSITE POWER DURING TESTING	PALISADES	196
097107	741022	MSLB	BIT AUTO INJECTION FAILURE	ZION 2	192
097779	741124	RCPT	FAILURE OF RPS TO SEE RCP TRIPS	ARKANSAS 1	200
093553	741208	LOOP	TWO DIESEL GENERATORS MALFUNCTION	HATCH 1	179
100415	750225	LOFW	ALL DRYWELL HIGH PRESSURE SWITCHES ISOLATED	FITZPATRIK	205
101078	750318	LOOP	DIESEL GENERATOR FIRE	FITZPATRIK	209
101444	750322	LOFW	CABLE TRAY FIRE CAUSED EXTENSIVE DAMAGE	BRN.FERRY1	213
102146	750320	LOOP	LOSS OF OFFSITE POWER	BRUNSWICK2	220
103111	750419	LOOP	CONDENSER EMERG. DISCHARGE VLVS FAIL TO OPEN	OCONEE 3	233
103002	750429	LOFW	MULTIPLE VALVE FAILURES & RCIC INOPERABLE	BRUNSWICK2	224
103077	750501	LOCA	RCP SEAL FAILURE	ROBINSON 2	228
103207	750525	LOOP	BREAKER & RELAY FAILURE DURING TEST	TKY.POINT4	237
103892	750613	LOCA	STUCK OPEN PORV	OCONEE 3	241
104087	750619	LOCA	CLOSED SI TANK ISOLATION VALVES	MA.YANKEE	245
104379	750627	EQUK	EARTHQUAKE OCCURS	HUMBDLTBAY	249
104817	750729	INAA	ALL ADS VALVES OPEN INADVERTENTLY	HATCH 1	253
105540	750817	LOCA	LOW FLOW FEEDWATER LINE SEVERED AT 6X4 REDUCER	QUAD-CTES2	257
106333	750909	LOFW	HPCI FOUND INOPERABLE	QUAD-CTES2	261
106616	750913	LOOP	LOSS OF OFFSITE PWR & RELF.VALV STKS OPN	PILGRIM 1	265
108078	751105	LOFW	INOPERABLE AFW PUMPS DURING PLANT STARTUP	KEWAUNEE	269
109295	751219	LOCA	FROZEN BWST LEVEL INPULSE LINES WHILE OPRTING	OCONEE 1	277
108869	751220	LOOP	TWO DIESELS FAIL TO START	HATCH 1	273
112309	760308	LOOP	GAS TURBINE BECOMES INOPERABLE	MILLSTONE1	281
112310	760315	LOOP	GAS TURBINE FAILS TO ACHIEVE RATED SPEED	MILLSTONE1	285
115875	760626	LOOP	LOSS OF OFFSITE POWER DURING REFUELING	HAD.NECK	293
115870	760706	LOCA	IMPROPER INSULATION	VT.YANKEE	289
116212	760720	LOOP	APPARENT LOOP & FAILURE OF SAFETY RELATED COMP	MILLSTONE2	297
116780	760810	LOOP	GAS TURBINE FAILS DURING PLANT TRIP	MILLSTONE1	302
117944	760810	LOOP	REACTOR TRIP WITH LOOP & DIESEL GENS.RUNNING	MILLSTONE2	306
120207	761028	MSLB	FAILURE OF RWST-CHARGING PUMP SUCTION VALVES	TROJAN	310
120439	761101	LOFW	TWO ELECTROMATIC RV FAILURES	QUAD-CTES1	318
120443	761101	LOFW	TWO ELECTROMATIC RV FAILURES	QUAD-CTES2	323
120293	761111	LOOP	PLANT SERVICE WATER STRAINERS PLUGGED	HATCH 1	314
120995	761111	LOCA	VALVES FAIL TO OPEN FOR DECAY HEAT SYSTEM	RANCHOSECO	326
121533	770104	LOOP	TWO DIESEL GENERATORS TRIP	BRUNSWICK1	330
122366	770110	MSLB	SENSING LINES FROZEN	ARKANSAS 1	335
123118	770218	LOFW	BOTH AFW PUMPS UNAVAILABLE	RANCHOSECO	339
123150	770303	MSLB	50% OPEN ATMOSPHERIC DUMP VALVES	CRYSTALRV3	343
124201	770322	LOOP	DIESEL GENERATOR FIELD FAILS TO FLASH	DRESDEN 3	347
124222	770408	LOFW	SIX MAIN STEAM RELIEF VALVES FAIL TO LIFT	D.ARNOLD	351
124920	770416	MSLB	STEAM GENERATOR OVERFILL	CRYSTALRV3	355
125106	770506	LOCA	FAILURE TO CLEAR OVERPRESSURE DURING HEATUP	SALEM 1	359
125563	770516	LOOP	REACTOR TRIP WITH LOSS OF OFFSITE POWER	ST.LUCIE 1	363
132979	770515	LOOP	FUEL SUPPLY VALVES FOUND SHUT DURING INSPECT.	MILLSTONE2	436
133004	770613	LOOP	3/4 DIESELS INOPERABLE	PEACHBTOM3	440
127010	770714	LOOP	FAILURE OF TWO DIESEL GENERATORS	DRESDEN 3	367
127384	770712	LOCA	DUMMY SIGNALS INSTALLED ON INSTRUMENTENTATION	ZION 2	371

Table B.1 (continued)

128569	770715	LOCA	SAFETY RELIEF VALVE FAILS TO RESET	BRUNSWICK2	376
143219	770713	LOOP	LOSS OF OFFSITE POWER	IND.POINT3	545
143255	770730	LOOP	DIESEL GENERATORS FAILED TO START AT TEST	SALEM 1	549
128935	770822	LOOP	MOMENTARY LOSS OF OFFSITE POWER	FTCALHOUN1	386
131225	770828	LOOP	TWO DIESEL GENERATORS FAIL TO START	PEACHBTOM2	407
128906	770831	LOFW	LOSS OF NO-BREAK-POWER AND FEEDWATER CONTROL	COOPER	381
130111	770901	LOOP	REACTOR TRIP & LOSS OF OFFSITE POWER	COOK 1	391
135006	770916	LOOP	LOSS OF OFFSITE POWER	FARLEY 1	449
130119	770924	LOOP	REACTOR TRIP DUE TO LOSS OF OFFSITE POWER	PALISADES	395
130788	770924	LOCA	STUCK OPEN PORV	DVS-BESSE1	399
130889	771011	LOCA	FLOODING OF VALVE PIT DURING 0 PWR PHYSIC TST	SURRY 2	403
143533	771010	LOOP	DIESEL GENERATOR LOAD CYCLES	BRUNSWICK1	557
135077	771026	MSLB	THERMO WELL COUPLING FAILED AFTER SAFETY INJ.	ROBINSON 2	454
143296	771020	LOCA	INADVERTENT VALVE CLOSING DURING TEST	MA.YANKEE	553
132155	771129	LOOP	DIESEL GENERATOR TRIPS DURING TESTING	DRESDEN 3	411
132927	771129	LOOP	LOSS OF OFFSITE POWER	DVS-BESSE1	415
132943	771125	LOOP	REACTOR TRIP WITH LOSS OF OFFSITE POWER	PALISADES	424
135079	771123	LOCA	SI PUMP BREAKERS INCORRECTLY RACKED OUT	ROBINSON 2	458
132942	771209	LOOP	BOTH DIESEL GENERATORS INOPERABLE	PRAIRIEIS1	419
132947	771202	LOOP	DIESEL GENERATOR FAILS TO START	DRESDEN 2	428
132958	771211	LOOP	REACTOR TRIP WITH LOSS OF OFFSITE POWER	PALISADES	432
133706	771211	LOFW	AUX. FEEDWATER PUMPS INOPERABLE DURING TEST	DVS-BESSE1	445
144187	771210	LOOP	DIESEL GENERATOR FOUND INOPERABLE	MILLSTONE1	561
135932	780206	LOOP	LOSS OF OFFSITE POWER	PILGRIM 1	462
137384	780318	LOCA	RELAY FAILURE	TMI 1	474
137305	780325	LOFW	LOW-LOW WATER LEVEL IN ONE STEAM GENERATOR	FARLEY 1	466
138830	780320	LOFW	FAILURE OF NNI & STEAM GENERATOR DRYOUT	RANCHOSECO	486
141097	780329	LOCA	STUCK OPEN PORV	TMI 2	520
137335	780404	LOFW	IC INADVERNTLY RENDERED INOPERABLE	DRESDEN 2	470
137543	780413	LOOP	LOSS OF OFFSITE POWER WHILE SHUTDOWN	CALCLIFFS1	478
137918	780423	MSLB	MULTIPLE STUCK-OPEN RELIEF VALVES	TMI 2	482
139565	780514	LOOP	LOSS OF OFFSITE POWER DURING REFUELING	ST.LUCIE 1	491
139573	780615	LOOP	UNAVAILABILITY OF BOTH DIESEL GENERATORS	COOK 2	495
140335	780728	LOOP	LOOP AND DIESEL GENERATOR FAILURE	BVRVALLEY1	508
140140	780806	LOOP	LOSS OF OFFSITE POWER	PILGRIM 1	499
140162	780824	LOOP	DIESEL GENERATOR FAILS TO START	DRESDEN 2	504
140691	780905	LOOP	BOTH DIESEL GENERATORS OUT OF SERVICE	BVRVALLEY1	512
140693	780912	LOOP	BOTH DIESEL GENERATORS OUT OF SERVICE	BVRVALLEY1	516
141455	781004	LOOP	BOTH DIESEL GENERATORS OUT OF SERVICE	RANCHOSECO	524
141523	781017	LOCA	INOPERABLE LPSI PUMPS	CALCLIFFS2	528
141706	781024	LOOP	LOSS OF 1 DG WHILE 2ND OUT OF SERVICE	RANCHOSECO	532
142329	781126	MSLB	HEAT TRACING CIRCUIT FAILURE	IND.POINT2	536
142462	781127	LOCA	LOSS OF VITAL INST. BUS-REACTOR TRIP	SALEM 1	540
145209	781220	LOCA	TWO STUCK OPEN PROV'S DURING COLD SHUTDOWN	FTCALHOUN1	565
147223	790106	MSLB	TURBINE TRIP AND STUCK FW VALVE	CRYSTALRV3	578
146540	790110	LOOP	ONE DG FAILED WHILE 2ND OUT DUE TO MAINTENANCE	KEWAUNEE	569
146744	790112	LOFW	LOSS OF VITAL BUS WHILE AT POWER	DVS-BESSE1	573
148764	790118	MSLB	STUCK OPEN STEAM DUMP VALVES	BVRVALLEY1	595
147400	790219	MSLB	MAIN STEAM STOP VALVES FAILED TO CLOSE AT TEST	PT.BEACH 2	591
147383	790223	LOOP	TWO DIESEL GENERATORS NOT AVAILABLE (TWICE)	COOK 1	582
147399	790221	LOOP	BOTH DIESEL GENERATORS UNAVAILABLE	KEWAUNEE	587
148892	790222	LOOP	UNAVAILABILITY OF TWO DIESEL GENERATORS	FARLEY 1	599
149230	790321	LOOP	BOTH EMERGENCY SERVICE WATER SYSTEM OUT	D.ARNOLD	603
153164	790328	LOFW	LOSS OF FEEDWATER & OPEN PORV	TMI 2	688
149250	790414	MSLB	MSIVS FAILED TO CLOSE DURING TEST	TROJAN	607
149450	790502	LOFW	LOSS OF FEEDWATER FLOW	OYSTER CRK	611
149962	790505	LOOP	DIESEL GENERATOR UNABLE TO CARRY LOAD	HATCH 1	620
149961	790603	LOFW	HPCI FAILS TO START GIVEN LOFW	HATCH 1	616
150271	790603	LOFW	BOTH RHR SERVICE WATER PMPS NOT OPERABLE	BRUNSWICK2	628
150882	790606	LOOP	BOTH DG'S TRIPPED DURING TEST	CRYSTALRV3	640
150051	790612	LOOP	FAILURE OF TWO DIESEL GENERATORS	DRESDEN 2	624
150499	790628	LOFW	RCIC AND HPCI ISOLATE FOLLOWING REACTOR SCRAM	HATCH 2	632
150742	790622	LOFW	MAIN STACK SAMPLING SYSTEM LOST	PEACHBTOM3	636
150943	790727	LOOP	LOSS OF OFFSITE POWER	PILGRIM 1	644
152835	790731	LOOP	OVR VOLTG PROT TRIP FOUND WHILE TRBLE SHTG	CALCLIFFS1	675
152183	790813	LOCA	OPEN PWR OPERATED RELIEF VALVE + ISO.VALVE	HAD.NECK	658
151635	790828	LOOP	LOSS OF OFFSITE POWER	PILGRIM 1	653
152187	790903	LOOP	SWITCHYARD LOCKOUT DUE TO CABLE DROP AT STORM	ST.LUCIE 1	662
151433	790911	LOCA	INCORRECT RHR SERVICE PUMP BREAKER RACKED OUT	BRUNSWICK2	649
153167	790926	LOOP	LOSS OF STANDBY POWER SOURCE DURING PWR OPRT	OCONEE 1	693
153167	790926	LOOP	LOSS OF STANDBY POWER SOURCE DURING PWR OPRT	OCONEE 2	693
152563	791002	SGTR	STEAM GENERATOR TUBE BREAK	PRAIRIEIS1	666

Table B.1 (continued)

152638	791017	LOOP	UNAVAIL OF 1 AFW PUMP AND 1 DIESEL GENERATOR	COOK 2	671
152951	791015	LOOP	REACTOR TRIP WITH LOSS OF OFFSITE POWER	DVS-BESSE1	679
153003	791024	LOCA	FAILURE TO PROPERLY POSITION VALVES AFTER TEST	OCONEE 2	684
153338	791101	LOOP	INCORRECT DIESEL GENERATOR MAINT. WHILE REFUEL	COOK 2	701
153333	791115	LOFW	AFW UNAVAIL DUE TO MAINTENANCE AND INSPECTION	COOK 1	697
153686	791127	LOOP	DG AND CHARGING PMP SUCT VALVE UNAVAILABLE	BVRVALLEY1	705
153810	791120	LOFW	RCIC TURBINE TRIP WITH HPCI UNAVAILABLE	BRUNSWICK1	709
154286	791128	LOCA	ONE PMP FAILED + 1 DEGRADED IN CLD SHTDN	SALEM 1	713
154639	791209	LOOP	BOTH DIESEL GENERATORS INOPERABLE	DVS-BESSE1	717

FILE SORTED BY SEQUENCE OF INTEREST AND ACCESSION NUMBER

ACCESS	E DATE	SEQ	ACTUAL OCCURRENCE	PLANT	PG
039380	711208	ECIT	SAFETY VALVE OPERATION AFTER LOFW	DRESDEN 3	25
047814	700505	ECIT	DEPRESSURIZATION INCIDENT	DRESDEN 2	33
074242	711231	ECIT	HIGH COOLANT LEVEL	NINEMIPT1	97
104379	750627	EQUK	EARTHQUAKE OCCURS	HUMBDLTBAY	249
104817	750729	INAA	ALL ADS VALVES OPEN INADVERTENTLY	HATCH 1	253
059484	710112	LOCA	SUMP ISO. VALVES CLOSED	PT.BEACH 1	37
065757	710416	LOCA	FAILURE OF CONTAINMENT SUMP ISO VALVES	PT.BEACH 1	75
065969	710908	LOCA	OPEN ELECTROMATIC RELIEF VALVE	PALISADES	79
066996	711010	LOCA	TRANSIENT AND BLOWDOWN	MILLSTONE1	84
077916	721229	LOCA	MALFUNCTION OF SEVERAL VALVES	OYSTER CRK	105
078418	730202	LOCA	VALVE FAILURE TO OPEN DURING TESTING	MA.YANKEE	110
103077	750501	LOCA	RCP SEAL FAILURE	ROBINSON 2	228
103892	750613	LOCA	STUCK OPEN PORV	OCONEE 3	241
104087	750619	LOCA	CLOSED SI TANK ISOLATION VALVES	MA.YANKEE	245
105540	750817	LOCA	LOW FLOW FEEDWATER LINE SEVERED AT 6X4 REDUCER	QUAD-CTES2	257
109295	751219	LOCA	FROZEN BWST LEVEL INPULSE LINES WHILE OPRTING	OCONEE 1	277
115870	760706	LOCA	IMPROPER INSULATION	VT.YANKEE	289
120995	761111	LOCA	VALVES FAIL TO OPEN FOR DECAY HEAT SYSTEM	RANCHOSECO	326
125106	770506	LOCA	FAILURE TO CLEAR OVERPRESSURE DURING HEATUP	SALEM 1	359
127384	770712	LOCA	DUMMY SIGNALS INSTALLED ON INSTRUMENTENTATION	ZION 2	371
128569	770715	LOCA	SAFETY RELIEF VALVE FAILS TO RESET	BRUNSWICK2	376
130788	770924	LOCA	STUCK OPEN PORV	DVS-BESSE1	399
130889	771011	LOCA	FLOODING OF VALVE PIT DURING 0 PWR PHYSIC TST	SURRY 2	403
135079	771123	LOCA	SI PUMP BREAKERS INCORRECTLY RACKED OUT	ROBINSON 2	458
137384	780318	LOCA	RELAY FAILURE	TMI 1	474
141097	780329	LOCA	STUCK OPEN PORV	TMI 2	520
141523	781017	LOCA	INOPERABLE LPSI PUMPS	CALCLIFFS2	528
142462	781127	LOCA	LOSS OF VITAL INST. BUS-REACTOR TRIP	SALEM 1	540
143296	771020	LOCA	INADVERTENT VALVE CLOSING DURING TEST	MA.YANKEE	553
145209	781220	LOCA	TWO STUCK OPEN PROV'S DURING COLD SHUTDOWN	FTCALHOUN1	565
151433	790911	LOCA	INCORRECT RHR SERVICE PUMP BREAKER RACKED OUT	BRUNSWICK2	649
152183	790813	LOCA	OPEN PWR OPERATED RELIEF VALVE + ISO.VALVE	HAD.NECK	658
153003	791024	LOCA	FAILURE TO PROPERLY POSITION VALVES AFTER TEST	OCONEE 2	684
154286	791128	LOCA	ONE PMP FAILED + 1 DEGRADED IN CLD SHTDN	SALEM 1	713
044751	700427	LOFW	3 OF 4 STEAM DRUM SCRAM SENSORS FAIL	DRESDEN 1	29
081523	730618	LOFW	FAILURE OF AFW PUMPS TO AUTO-START	TKY.POINT4	123
085738	731119	LOFW	RCIC/HPCI FAILS DURING TESTING	BRN.FERRY1	147
090421	740407	LOFW	INOPERABLE AFW PUMPS DURING PLANT SHUTDOWN	PT.BEACH 1	171
091676	740508	LOFW	FAILURE OF 3 AUX FWDTR PMPS TO START AT TEST	TKY.POINT3	175
095040	740823	LOFW	HPCI INJECTION VALVE AND RCIC FAIL	BRN.FERRY2	188
100415	750225	LOFW	ALL DRYWELL HIGH PRESSURE SWITCHES ISOLATED	FITZPATRIK	205
101444	750322	LOFW	CABLE TRAY FIRE CAUSED EXTENSIVE DAMAGE	BRN.FERRY1	213
103002	750429	LOFW	MULTIPLE VALVE FAILURES & RCIC INOPERABLE	BRUNSWICK2	224
106333	750909	LOFW	HPCI FOUND INOPERABLE	QUAD-CTES2	261
108078	751105	LOFW	INOPERABLE AFW PUMPS DURING PLANT STARTUP	KEWAUNEE	269
120439	761101	LOFW	TWO ELECTROMATIC RV FAILURES	QUAD-CTES1	318
120443	761101	LOFW	TWO ELECTROMATIC RV FAILURES	QUAD-CTES2	323
123118	770218	LOFW	BOTH AFW PUMPS UNAVAILABLE	RANCHOSECO	339
124222	770408	LOFW	SIX MAIN STEAM RELIEF VALVES FAIL TO LIFT	D.ARNOLD	351
128906	770831	LOFW	LOSS OF NO-BREAK-POWER AND FEEDWATER CONTROL	COOPER	381
133706	771211	LOFW	AUX. FEEDWATER PUMPS INOPERABLE DURING TEST	DVS-BESSE1	445
137305	780325	LOFW	LOW-LOW WATER LEVEL IN ONE STEAM GENERATOR	FARLEY 1	466
137335	780404	LOFW	IC INADVERNTLY RENDERED INOPERABLE	DRESDEN 2	470
138830	780320	LOFW	FAILURE OF NNI & STEAM GENERATOR DRYOUT	RANCHOSECO	486
146744	790112	LOFW	LOSS OF VITAL BUS WHILE AT POWER	DVS-BESSE1	573
149450	790502	LOFW	LOSS OF FEEDWATER FLOW	OYSTER CRK	611
149961	790603	LOFW	HPCI FAILS TO START GIVEN LOFW	HATCH 1	616
150271	790603	LOFW	BOTH RHR SERVICE WATER PMPS NOT OPERABLE	BRUNSWICK2	628

Table B.1 (continued)

150499	730628	LOFW	RCIC AND HPCI ISOLATE FOLLOWING REACTOR SCRAM	HATCH 2	632
150742	790622	LOFW	MAIN STACK SAMPLING SYSTEM LOST	PEACHBTOM3	636
153164	790328	LOFW	LOSS OF FEEDWATER & OPEN PORV	TMI 2	688
153333	791115	LOFW	AFW UNAVAIL DUE TO MAINTENANCE AND INSPECTION	COOK 1	697
153810	791120	LOFW	RCIC TURBINE TRIP WITH HPCI UNAVAILABLE	BRUNSWICK1	709
036147	690715	LOOP	REACTOR TRIP WITH LOSS OF OFFSITE POWER	HAD.NECK	16
039024	720125	LOOP	LOSS OF OFFSITE POWER	BG ROCK PT	20
061043	710120	LOOP	LOSS OF OFFSITE POWER	LACROSSE	45
061434	700717	LOOP	LOSS OF OFFSITE POWER	HUMBDLTBAY	49
061565	710902	LOOP	LOOP AND FAILURE OF A DIESEL GENERATOR TO LOAD	PALISADES	54
063129	710324	LOOP	LOSS OF OFFSITE POWER	LACROSSE	58
063144	710308	LOOP	FAILURE OF BOTH DIESEL GENER. DURING TESTING	ROBINSON 2	63
064600	701226	LOOP	SHUTDOWN OF BUCHANAN STATION	IND.POINT1	67
065757	710205	LOOP	LOSS OF OFFSITE POWER	PT.BEACH 1	71
071694	720517	LOOP	LOSS OF OFFSITE POWER	PALISADES	89
073655	720720	LOOP	LOSS OF OFFSITE POWER	IND.POINT1	93
075074	720817	LOOP	LOSS OF LOAD	LACROSSE	101
079565	721201	LOOP	LOSS OF NORMAL STATION POWER	VT.YANKEE	114
080138	720610	LOOP	FLOODING OF TURBINE BUILDING	QUAD-CTES1	118
082198	730704	LOOP	REACTOR/TURBINE TRIP DURING TEST	TKY.POINT3	127
083833	730908	LOOP	LOSS OF OFFSITE POWER	OYSTER CRK	131
085370	731021	LOCP	LOOP,EXCESSIVE RCS COOLDOWN,SFTY INJ.&INS BUS	GINNA	137
085566	731119	LOOP	LOSS OF A.C. POWER CAUSE HPCI/RCIC TO FAIL	BRN.FERRY1	143
086990	731121	LOOP	WORKER BUMPS RELAY CAUSING LOOP	NINEMIPT1	151
088451	740119	LOOP	LOSS OF OFFSITE POWER	HAD.NECK	155
089252	740217	LOOP	FAILURE OF DIESEL TO START/LOAD;2 AFW PMPS OUT	ZION 2	167
093553	741208	LOOP	TWO DIESEL GENERATORS MALFUNCTION	HATCH 1	179
093702	740425	LOOP	REACTOR TRIP WITH LOSS OF OFFSITE POWER	TKY.POINT4	183
097578	741017	LOOP	LOSS OF OFFSITE POWER DURING TESTING	PALISADES	196
101078	750318	LOOP	DIESEL GENERATOR FIRE	FITZPATRIK	209
102146	750320	LOOP	LOSS OF OFFSITE POWER	BRUNSWICK2	220
103111	750419	LOOP	CONDENSER EMERG. DISCHARGE VLVS FAIL TO OPEN	OCONEE 3	233
103207	750525	LOOP	BREAKER & RELAY FAILURE DURING TEST	TKY.POINT4	237
106616	750913	LOOP	LOSS OF OFFSITE PWR & RELF.VALV STKS OPN	PILGRIM 1	265
108869	751220	LOOP	TWO DIESELS FAIL TO START	HATCH 1	273
112309	760308	LOOP	GAS TURBINE BECOMES INOPERABLE	MILLSTONE1	281
112310	760315	LOOP	GAS TURBINE FAILS TO ACHIEVE RATED SPEED	MILLSTONE1	285
115875	760626	LOOP	LOSS OF OFFSITE POWER DURING REFUELING	HAD.NECK	293
116212	760720	LOOP	APPARENT LOOP & FAILURE OF SAFETY RELATED COMP	MILLSTONE2	297
116780	760810	LOOP	GAS TURBINE FAILS DURING PLANT TRIP	MILLSTONE1	302
117944	760810	LOOP	REACTOR TRIP WITH LOOP & DIESEL GENS.RUNNING	MILLSTONE2	306
120293	761111	LOOP	PLANT SERVICE WATER STRAINERS PLUGGED	HATCH 1	314
121533	770104	LOOP	TWO DIESEL GENERATORS TRIP	BRUNSWICK1	330
124201	770322	LOOP	DIESEL GENERATOR FIELD FAILS TO FLASH	DRESDEN 3	347
125563	770516	LOOP	REACTOR TRIP WITH LOSS OF OFFSITE POWER	ST.LUCIE 1	363
127010	770714	LOOP	FAILURE OF TWO DIESEL GENERATORS	DRESDEN 3	367
128935	770822	LOOP	MOMENTARY LOSS OF OFFSITE POWER	FTCALHOUN1	386
130111	770901	LOOP	REACTOR TRIP & LOSS OF OFFSITE POWER	COOK 1	391
130119	770924	LOOP	REACTOR TRIP DUE TO LOSS OF OFFSITE POWER	PALISADES	395
131225	770828	LOOP	TWO DIESEL GENERATORS FAIL TO START	PEACHBTOM2	407
132155	771129	LOOP	DIESEL GENERATOR TRIPS DURING TESTING	DRESDEN 3	411
132927	771129	LOOP	LOSS OF OFFSITE POWER	DVS-BESSE1	415
132942	771209	LOOP	BOTH DIESEL GENERATORS INOPERABLE	PRAIRIEIS1	419
132943	771125	LOOP	REACTOR TRIP WITH LOSS OF OFFSITE POWER	PALISADES	424
132947	771202	LOOP	DIESEL GENERATOR FAILS TO START	DRESDEN 2	428
132958	771211	LOOP	REACTOR TRIP WITH LOSS OF OFFSITE POWER	PALISADES	432
132979	770515	LOOP	FUEL SUPPLY VALVES FOUND SHUT DURING INSPECT.	MILLSTONE2	436
133004	770613	LOOP	3/4 DIESELS INOPERABLE	PEACHBTOM3	440
135006	770916	LOOP	LOSS OF OFFSITE POWER	FARLEY 1	449
135932	780206	LOOP	LOSS OF OFFSITE POWER	PILGRIM 1	462
137543	780413	LOOP	LOSS OF OFFSITE POWER WHILE SHUTDOWN	CALCLIFFS1	478
139565	780514	LOOP	LOSS OF OFFSITE POWER DURING REFUELING	ST.LUCIE 1	491
139573	780615	LOOP	UNAVAILABILITY OF BOTH DIESEL GENERATORS	COOK 2	495
140140	780806	LOOP	LOSS OF OFFSITE POWER	PILGRIM 1	499
140162	780824	LOOP	DIESEL GENERATOR FAILS TO START	DRESDEN 2	504
140335	780728	LOOP	LOOP AND DIESEL GENERATOR FAILURE	BVRVALLEY1	508
140691	780905	LOOP	BOTH DIESEL GENERATORS OUT OF SERVICE	BVRVALLEY1	512
140693	780912	LOOP	BOTH DIESEL GENERATORS OUT OF SERVICE	BVRVALLEY1	516
141455	781004	LOOP	BOTH DIESEL GENERATORS OUT OF SERVICE	RANCHOSECO	524
141706	781024	LOOP	LOSS OF 1 DG WHILE 2ND OUT OF SERVICE	RANCHOSECO	532
143219	770713	LOOP	LOSS OF OFFSITE POWER	IND.POINT3	545
143255	770730	LOOP	DIESEL GENERATORS FAILED TO START AT TEST	SALEM 1	549

Table B.1 (continued)

143533	771010	LOOP	DIESEL GENERATOR LOAD CYCLES	BRUNSWICK1	557
144187	771210	LOOP	DIESEL GENERATOR FOUND INOPERABLE	MILLSTONE1	561
146540	790110	LOOP	ONE DG FAILED WHILE 2ND OUT DUE TO MAINTENANCE	KEWAUNEE	569
147383	790223	LOOP	TWO DIESEL GENERATORS NOT AVAILABLE (TWICE)	COOK 1	582
147399	790221	LOOP	BOTH DIESEL GENERATORS UNAVAILABLE	KEWAUNEE	587
148892	790222	LOOP	UNAVAILABILITY OF TWO DIESEL GENERATORS	FARLEY 1	599
149230	790321	LOOP	BOTH EMERGENCY SERVICE WATER SYSTEM OUT	D.ARNOLD	603
149962	790505	LOOP	DIESEL GENERATOR UNABLE TO CARRY LOAD	HATCH 1	620
150051	790612	LOOP	FAILURE OF TWO DIESEL GENERATORS	DRESDEN 2	624
150882	790606	LOOP	BOTH DG'S TRIPPED DURING TEST	CRYSTALRV3	640
150943	790727	LOOP	LOSS OF OFFSITE POWER	PILGRIM 1	644
151635	790828	LOOP	LOSS OF OFFSITE POWER	PILGRIM 1	653
152187	790903	LOOP	SWITCHYARD LOCKOUT DUE TO CABLE DROP AT STORM	ST.LUCIE 1	662
152638	791017	LOOP	UNAVAIL OF 1 AFW PUMP AND 1 DIESEL GENERATOR	COOK 2	671
152835	790731	LOOP	OVR VOLTG PROT TRIP FOUND WHILE TRBLE SHTG	CALCLIFFS1	675
152951	791015	LOOP	REACTOR TRIP WITH LOSS OF OFFSITE POWER	DVS-BESSE1	679
153167	790926	LOOP	LOSS OF STANDBY POWER SOURCE DURING PWR OPRT	OCONEE 1	693
153167	790926	LOOP	LOSS OF STANDBY POWER SOURCE DURING PWR OPRT	OCONEE 2	693
153338	791101	LOOP	INCORRECT DIESEL GENERATOR MAINT. WHILE REFUEL	COOK 2	701
153686	791127	LOOP	DG AND CHARGING PMP SUCT VALVE UNAVAILABLE	BVRVALLEY1	705
154639	791209	LOOP	BOTH DIESEL GENERATORS INOPERABLE	DVS-BESSE1	717
089205	731130	LRTR	FAILED RCP SHAFT AT 92% POWER	SURRY 1	163
032429	690313	MSLB	LOOP ISOLAT. VALVES FAILED TO SHUT	IND.POINT1	12
060227	701204	MSLB	FAILURE OF SEVERAL MSIV'S TO CLOSE	DRESDEN 2	41
089153	740221	MSLB	OPEN RWST ISOLATION VALVES	PRAIRIEIS1	159
097107	741022	MSLB	BIT AUTO INJECTION FAILURE	ZION 2	192
120207	761028	MSLB	FAILURE OF RWST-CHARGING PUMP SUCTION VALVES	TROJAN	310
122366	770110	MSLB	SENSING LINES FROZEN	ARKANSAS 1	335
123150	770303	MSLB	50% OPEN ATMOSPHERIC DUMP VALVES	CRYSTALRV3	343
124920	770416	MSLB	STEAM GENERATOR OVERFILL	CRYSTALRV3	355
135077	771026	MSLB	THERMO WELL COUPLING FAILED AFTER SAFETY INJ.	ROBINSON 2	454
137918	780423	MSLB	MULTIPLE STUCK-OPEN RELIEF VALVES	TMI 2	482
142329	781126	MSLB	HEAT TRACING CIRCUIT FAILURE	IND.POINT2	536
147223	790106	MSLB	TURBINE TRIP AND STUCK FW VALVE	CRYSTALRV3	578
147400	790219	MSLB	MAIN STEAM STOP VALVES FAILED TO CLOSE AT TEST	PT.BEACH 2	591
148764	790118	MSLB	STUCK OPEN STEAM DUMP VALVES	BVRVALLEY1	595
149250	790414	MSLB	MSIVS FAILED TO CLOSE DURING TEST	TROJAN	607
097779	741124	RCPT	FAILURE OF RPS TO SEE RCP TRIPS	ARKANSAS 1	200
152563	791002	SGTR	STEAM GENERATOR TUBE BREAK	PRAIRIEIS1	666

FILE SORTED BY PLANT NAME AND ACCESSION NUMBER (ONLY PLANTS WITH PRECURSORS INCLUDED)

ACCESS	E DATE	SEQ	ACTUAL OCCURRENCE	PLANT	PG
097779	741124	RCPT	FAILURE OF RPS TO SEE RCP TRIPS	ARKANSAS 1	200
122366	770110	MSLB	SENSING LINES FROZEN	ARKANSAS 1	335
039024	720125	LOOP	LOSS OF OFFSITE POWER	BG ROCK PT	20
085566	731119	LOOP	LOSS OF A.C. POWER CAUSE HPCI/RCIC TO FAIL	BRN.FERRY1	143
085738	731119	LOFW	RCIC/HPCI FAILS DURING TESTING	BRN.FERRY1	147
095040	740823	LOFW	HPCI INJECTION VALVE AND RCIC FAIL	BRN.FERRY2	188
101444	750322	LOFW	CABLE TRAY FIRE CAUSED EXTENSIVE DAMAGE	BRN.FERRY1	213
102146	750320	LOOP	LOSS OF OFFSITE POWER	BRUNSWICK2	220
103002	750429	LOFW	MULTIPLE VALVE FAILURES & RCIC INOPERABLE	BRUNSWICK2	224
121533	770104	LOOP	TWO DIESEL GENERATORS TRIP	BRUNSWICK1	330
128569	770715	LOCA	SAFETY RELIEF VALVE FAILS TO RESET	BRUNSWICK2	376
143533	771010	LOOP	DIESEL GENERATOR LOAD CYCLES	BRUNSWICK1	557
150271	790603	LOFW	BOTH RHR SERVICE WATER PMPS NOT OPERABLE	BRUNSWICK2	628
151433	790911	LOCA	INCORRECT RHR SERVICE PUMP BREAKER RACKED OUT	BRUNSWICK2	649
153810	791120	LOFW	RCIC TURBINE TRIP WITH HPCI UNAVAILABLE	BRUNSWICK1	709
140335	780728	LOOP	LOOP AND DIESEL GENERATOR FAILURE	BVRVALLEY1	508
140691	780905	LOOP	BOTH DIESEL GENERATORS OUT OF SERVICE	BVRVALLEY1	512
140693	780912	LOOP	BOTH DIESEL GENERATORS OUT OF SERVICE	BVRVALLEY1	516
148764	790118	MSLB	STUCK OPEN STEAM DUMP VALVES	BVRVALLEY1	595
153686	791127	LOOP	DG AND CHARGING PMP SUCT VALVE UNAVAILABLE	BVRVALLEY1	705
137543	780413	LOOP	LOSS OF OFFSITE POWER WHILE SHUTDOWN	CALCLIFFS1	478
141523	781017	LOCA	INOPERABLE LPSI PUMPS	CALCLIFFS2	528
152835	790731	LOOP	OVR VOLTG PROT TRIP FOUND WHILE TRBLE SHTG	CALCLIFFS1	675
130111	770901	LOOP	REACTOR TRIP & LOSS OF OFFSITE POWER	COOK 1	391
147383	790223	LOOP	TWO DIESEL GENERATORS NOT AVAILABLE (TWICE)	COOK 1	582
153333	791115	LOFW	AFW UNAVAIL DUE TO MAINTENANCE AND INSPECTION	COOK 1	697
139573	780615	LOOP	UNAVAILABILITY OF BOTH DIESEL GENERATORS	COOK 2	495
152638	791017	LOOP	UNAVAIL OF 1 AFW PUMP AND 1 DIESEL GENERATOR	COOK 2	671
153338	791101	LOOP	INCORRECT DIESEL GENERATOR MAINT. WHILE REFUEL	COOK 2	701

[redacted]

Table B.1 (continued)

123906	770831	LOFW	LOSS OF NO-BREAK-POWER AND FEEDWATER CONTROL	COOPER	381
123150	770303	MSLB	50% OPEN ATMOSPHERIC DUMP VALVES	CRYSTALRV3	343
124920	770416	MSLB	STEAM GENERATOR OVERFILL	CRYSTALRV3	355
147223	790106	MSLB	TURBINE TRIP AND STUCK FW VALVE	CRYSTALRV3	578
150882	790606	LOOP	BOTH DG'S TRIPPED DURING TEST	CRYSTALRV3	640
124222	770408	LOFW	SIX MAIN STEAM RELIEF VALVES FAIL TO LIFT	D.ARNOLD	351
149230	790321	LOOP	BOTH EMERGENCY SERVICE WATER SYSTEM OUT	D.ARNOLD	603
044751	700427	LOFW	3 OF 4 STEAM DRUM SCRAM SENSORS FAIL	DRESDEN 1	29
047814	700505	ECIT	DEPRESSURIZATION INCIDENT	DRESDEN 2	33
060227	701204	MSLB	FAILURE OF SEVERAL MSIV'S TO CLOSE	DRESDEN 2	41
132947	771202	LOOP	DIESEL GENERATOR FAILS TO START	DRESDEN 2	428
137335	780404	LOFW	IC INADVERNTLY RENDERED INOPERABLE	DRESDEN 2	470
140162	780824	LOOP	DIESEL GENERATOR FAILS TO START	DRESDEN 2	504
150051	790612	LOOP	FAILURE OF TWO DIESEL GENERATORS	DRESDEN 2	624
039380	711208	ECIT	SAFETY VALVE OPERATION AFTER LOFW	DRESDEN 3	25
124201	770322	LOOP	DIESEL GENERATOR FIELD FAILS TO FLASH	DRESDEN 3	347
127010	770714	LOOP	FAILURE OF TWO DIESEL GENERATORS	DRESDEN 3	367
132155	771129	LOOP	DIESEL GENERATOR TRIPS DURING TESTING	DRESDEN 3	411
130788	770924	LOCA	STUCK OPEN PORV	DVS-BESSE1	399
132927	771129	LOOP	LOSS OF OFFSITE POWER	DVS-BESSE1	415
133706	771211	LOFW	AUX. FEEDWATER PUMPS INOPERABLE DURING TEST	DVS-BESSE1	445
146744	790112	LOFW	LOSS OF VITAL BUS WHILE AT POWER	DVS-BESSE1	573
152951	791015	LOOP	REACTOR TRIP WITH LOSS OF OFFSITE POWER	DVS-BESSE1	679
154639	791209	LOOP	BOTH DIESEL GENERATORS INOPERABLE	DVS-BESSE1	717
135006	770916	LOOP	LOSS OF OFFSITE POWER	FARLEY 1	449
137305	780325	LOFW	LOW-LOW WATER LEVEL IN ONE STEAM GENERATOR	FARLEY 1	466
148892	790222	LOOP	UNAVAILABILITY OF TWO DIESEL GENERATORS	FARLEY 1	599
100415	750225	LOFW	ALL DRYWELL HIGH PRESSURE SWITCHES ISOLATED	FITZPATRIK	205
101078	750318	LOOP	DIESEL GENERATOR FIRE	FITZPATRIK	209
128935	770822	LOOP	MOMENTARY LOSS OF OFFSITE POWER	FTCALHOUN1	386
145209	781220	LOCA	TWO STUCK OPEN PROV'S DURING COLD SHUTDOWN	FTCALHOUN1	565
085370	731021	LOOP	LOOP,EXCESSIVE RCS COOLDOWN,SFTY INJ.&INS BUS	GINNA	137
036147	690715	LOOP	REACTOR TRIP WITH LOSS OF OFFSITE POWER	HAD.NECK	16
088451	740119	LOOP	LOSS OF OFFSITE POWER	HAD.NECK	155
115875	760626	LOOP	LOSS OF OFFSITE POWER DURING REFUELING	HAD.NECK	293
152183	790813	LOCA	OPEN PWR OPERATED RELIEF VALVE + ISO.VALVE	HAD.NECK	658
093553	741208	LOOP	TWO DIESEL GENERATORS MALFUNCTION	HATCH 1	179
104817	750729	INAA	ALL ADS VALVES OPEN INADVERTENTLY	HATCH 1	253
108869	751220	LOOP	TWO DIESELS FAIL TO START	HATCH 1	273
120293	761111	LOOP	PLANT SERVICE WATER STRAINERS PLUGGED	HATCH 1	314
149961	790603	LOFW	HPCI FAILS TO START GIVEN LOFW	HATCH 1	616
149962	790505	LOOP	DIESEL GENERATOR UNABLE TO CARRY LOAD	HATCH 1	620
150499	790628	LOFW	RCIC AND HPCI ISOLATE FOLLOWING REACTOR SCRAM	HATCH 2	632
061434	700717	LOOP	LOSS OF OFFSITE POWER	HUMBDLTBAY	49
104379	750627	EQUK	EARTHQUAKE OCCURS	HUMBDLTBAY	249
032429	690313	MSLB	LOOP ISOLAT. VALVES FAILED TO SHUT	IND.POINT1	12
064600	701226	LOOP	SHUTDOWN OF BUCHANAN STATION	IND.POINT1	67
073655	720720	LOOP	LOSS OF OFFSITE POWER	IND.POINT1	93
142329	781126	MSLB	HEAT TRACING CIRCUIT FAILURE	IND.POINT2	536
143219	770713	LOOP	LOSS OF OFFSITE POWER	IND.POINT3	545
108078	751105	LOFW	INOPERABLE AFW PUMPS DURING PLANT STARTUP	KEWAUNEE	269
146540	790110	LOOP	ONE DG FAILED WHILE 2ND OUT DUE TO MAINTENANCE	KEWAUNEE	569
147399	790221	LOOP	BOTH DIESEL GENERATORS UNAVAILABLE	KEWAUNEE	587
061043	710120	LOOP	LOSS OF OFFSITE POWER	LACROSSE	45
063129	710324	LOOP	LOSS OF OFFSITE POWER	LACROSSE	58
075074	720817	LOOP	LOSS OF LOAD	LACROSSE	101
078418	730202	LOCA	VALVE FAILURE TO OPEN DURING TESTING	MA.YANKEE	110
104087	750619	LOCA	CLOSED SI TANK ISOLATION VALVES	MA.YANKEE	245
143296	771020	LOCA	INADVERTENT VALVE CLOSING DURING TEST	MA.YANKEE	553
066996	711010	LOCA	TRANSIENT AND BLOWDOWN	MILLSTONE1	84
112309	760308	LOOP	GAS TURBINE BECOMES INOPERABLE	MILLSTONE1	281
112310	760315	LOOP	GAS TURBINE FAILS TO ACHIEVE RATED SPEED	MILLSTONE1	285
116212	760720	LOOP	APPARENT LOOP & FAILURE OF SAFETY RELATED COMP	MILLSTONE2	297
116780	760810	LOOP	GAS TURBINE FAILS DURING PLANT TRIP	MILLSTONE1	302
117944	760810	LOOP	REACTOR TRIP WITH LOOP & DIESEL GENS.RUNNING	MILLSTONE2	306
132979	770515	LOOP	FUEL SUPPLY VALVES FOUND SHUT DURING INSPECT.	MILLSTONE2	436
144187	771210	LOOP	DIESEL GENERATOR FOUND INOPERABLE	MILLSTONE1	561
074242	711231	ECIT	HIGH COOLANT LEVEL	NINEMIPT1	97
086990	731121	LOOP	WORKER BUMPS RELAY CAUSING LOOP	NINEMIPT1	151
109295	751219	LOCA	FROZEN BWST LEVEL INPULSE LINES WHILE OPRTING	OCONEE 1	277
153167	790926	LOOP	LOSS OF STANDBY POWER SOURCE DURING PWR OPRT	OCONEE 1	693
153003	791024	LOCA	FAILURE TO PROPERLY POSITION VALVES AFTER TEST	OCONEE 2	684

Table B.1 (continued)

153167	790926	LOOP	LOSS OF STANDBY POWER SOURCE DURING PWR OPRT	OCONEE 2	693
103111	750419	LOOP	CONDENSER EMERG. DISCHARGE VLVS FAIL TO OPEN	OCONEE 3	233
103892	750613	LOCA	STUCK OPEN PORV	OCONEE 3	241
077916	721229	LOCA	MALFUNCTION OF SEVERAL VALVES	OYSTER CRK	105
083833	730908	LOOP	LOSS OF OFFSITE POWER	OYSTER CRK	131
149450	790502	LOFW	LOSS OF FEEDWATER FLOW	OYSTER CRK	611
061565	710902	LOOP	LOOP AND FAILURE OF A DIESEL GENERATOR TO LOAD	PALISADES	54
065969	710908	LOCA	OPEN ELECTROMATIC RELIEF VALVE	PALISADES	79
071694	720517	LOOP	LOSS OF OFFSITE POWER	PALISADES	89
097578	741017	LOOP	LOSS OF OFFSITE POWER DURING TESTING	PALISADES	196
130119	770924	LOOP	REACTOR TRIP DUE TO LOSS OF OFFSITE POWER	PALISADES	395
132943	771125	LOOP	REACTOR TRIP WITH LOSS OF OFFSITE POWER	PALISADES	424
132958	771211	LOOP	REACTOR TRIP WITH LOSS OF OFFSITE POWER	PALISADES	432
131225	770828	LOOP	TWO DIESEL GENERATORS FAIL TO START	PEACHBTOM2	407
133004	770613	LOOP	3/4 DIESELS INOPERABLE	PEACHBTOM3	440
150742	790622	LOFW	MAIN STACK SAMPLING SYSTEM LOST	PEACHBTOM3	636
106616	750913	LOOP	LOSS OF OFFSITE PWR & RELF.VALV STKS OPN	PILGRIM 1	265
135932	780206	LOOP	LOSS OF OFFSITE POWER	PILGRIM 1	462
140140	780806	LOOP	LOSS OF OFFSITE POWER	PILGRIM 1	499
150943	790727	LOOP	LOSS OF OFFSITE POWER	PILGRIM 1	644
151635	790828	LOOP	LOSS OF OFFSITE POWER	PILGRIM 1	653
089153	740221	MSLB	OPEN RWST ISOLATION VALVES	PRAIRIEIS1	159
132942	771209	LOOP	BOTH DIESEL GENERATORS INOPERABLE	PRAIRIEIS1	419
152563	791002	SGTR	STEAM GENERATOR TUBE BREAK	PRAIRIEIS1	666
059484	710112	LOCA	SUMP ISO. VALVES CLOSED	PT.BEACH 1	37
065757	710205	LOOP	LOSS OF OFFSITE POWER	PT.BEACH 1	71
065757	710416	LOCA	FAILURE OF CONTAINMENT SUMP ISO VALVES	PT.BEACH 1	75
090421	740407	LOFW	INOPERABLE AFW PUMPS DURING PLANT SHUTDOWN	PT.BEACH 1	171
147400	790219	MSLB	MAIN STEAM STOP VALVES FAILED TO CLOSE AT TEST	PT.BEACH 2	591
080138	720610	LOOP	FLOODING OF TURBINE BUILDING	QUAD-CTES1	118
105540	750817	LOCA	LOW FLOW FEEDWATER LINE SEVERED AT 6X4 REDUCER	QUAD-CTES2	257
106333	750909	LOFW	HPCI FOUND INOPERABLE	QUAD-CTES2	261
120439	761101	LOFW	TWO ELECTROMATIC RV FAILURES	QUAD-CTES1	318
120443	761101	LOFW	TWO ELECTROMATIC RV FAILURES	QUAD-CTES2	323
120995	761111	LOCA	VALVES FAIL TO OPEN FOR DECAY HEAT SYSTEM	RANCHOSECO	326
123118	770218	LOFW	BOTH AFW PUMPS UNAVAILABLE	RANCHOSECO	339
138830	780320	LOFW	FAILURE OF NNI & STEAM GENERATOR DRYOUT	RANCHOSECO	486
141455	781004	LOOP	BOTH DIESEL GENERATORS OUT OF SERVICE	RANCHOSECO	524
141706	781024	LOOP	LOSS OF 1 DG WHILE 2ND OUT OF SERVICE	RANCHOSECO	532
063144	710308	LOOP	FAILURE OF BOTH DIESEL GENER. DURING TESTING	ROBINSON 2	63
103077	750501	LOCA	RCP SEAL FAILURE	ROBINSON 2	228
135077	771026	MSLB	THERMO WELL COUPLING FAILED AFTER SAFETY INJ.	ROBINSON 2	454
135079	771123	LOCA	SI PUMP BREAKERS INCORRECTLY RACKED OUT	ROBINSON 2	458
125106	770506	LOCA	FAILURE TO CLEAR OVERPRESSURE DURING HEATUP	SALEM 1	359
142462	781127	LOCA	LOSS OF VITAL INST. BUS-REACTOR TRIP	SALEM 1	540
143255	770730	LOOP	DIESEL GENERATORS FAILED TO START AT TEST	SALEM 1	549
154286	791128	LOCA	ONE PMP FAILED + 1 DEGRADED IN CLD SHTDN	SALEM 1	713
125563	770516	LOOP	REACTOR TRIP WITH LOSS OF OFFSITE POWER	ST.LUCIE 1	363
139565	780514	LOOP	LOSS OF OFFSITE POWER DURING REFUELING	ST.LUCIE 1	491
152187	790903	LOOP	SWITCHYARD LOCKOUT DUE TO CABLE DROP AT STORM	ST.LUCIE 1	662
089205	731130	LRTR	FAILED RCP SHAFT AT 92% POWER	SURRY 1	163
130889	771011	LOCA	FLOODING OF VALVE PIT DURING 0 PWR PHYSIC TST	SURRY 2	403
081523	730618	LOFW	FAILURE OF AFW PUMPS TO AUTO-START	TKY.POINT4	123
082198	730704	LOOP	REACTOR/TURBINE TRIP DURING TEST	TKY.POINT3	127
091676	740508	LOFW	FAILURE OF 3 AUX FWDTR PMPS TO START AT TEST	TKY.POINT3	175
093702	740425	LOOP	REACTOR TRIP WITH LOSS OF OFFSITE POWER	TKY.POINT4	183
103207	750525	LOOP	BREAKER & RELAY FAILURE DURING TEST	TKY.POINT4	237
137384	780318	LOCA	RELAY FAILURE	TMI 1	474
137918	780423	MSLB	MULTIPLE STUCK-OPEN RELIEF VALVES	TMI 2	482
141097	780329	LOCA	STUCK OPEN PORV	TMI 2	520
153164	790328	LOFW	LOSS OF FEEDWATER & OPEN PORV	TMI 2	688
120207	761028	MSLB	FAILURE OF RWST-CHARGING PUMP SUCTION VALVES	TROJAN	310
149250	790414	MSLB	MSIVS FAILED TO CLOSE DURING TEST	TROJAN	607
079565	721201	LOOP	LOSS OF NORMAL STATION POWER	VT.YANKEE	114
115870	760706	LOCA	IMPROPER INSULATION	VT.YANKEE	289
089252	740217	LOOP	FAILURE OF DIESEL TO START/LOAD;2 AFW PMPS OUT	ZION 2	167
097107	741022	MSLB	BIT AUTO INJECTION FAILURE	ZION 2	192
127384	770712	LOCA	DUMMY SIGNALS INSTALLED ON INSTRUMENTENTATION	ZION 2	371

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 32429

Date: March 13, 1969

Title: Loop Isolation Valve Failure Due to Improper Lubrication at Indian Point 1

The failure sequence was:

1. Valve motor operators were improperly lubricated.
2. Several primary loop isolation valves failed to close during testing.

Corrective action;

1. A better grease was applied in an inspection and relubrication program.

Design purpose of failed system or component:

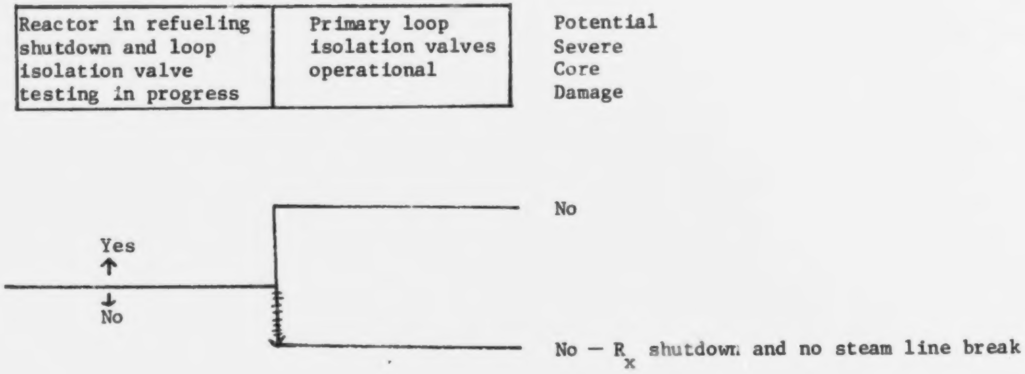
1. The valves serve to mitigate the consequences of steam line break accidents.

Unavailability of system per WASH 1400:* not considered in WASH-1400.

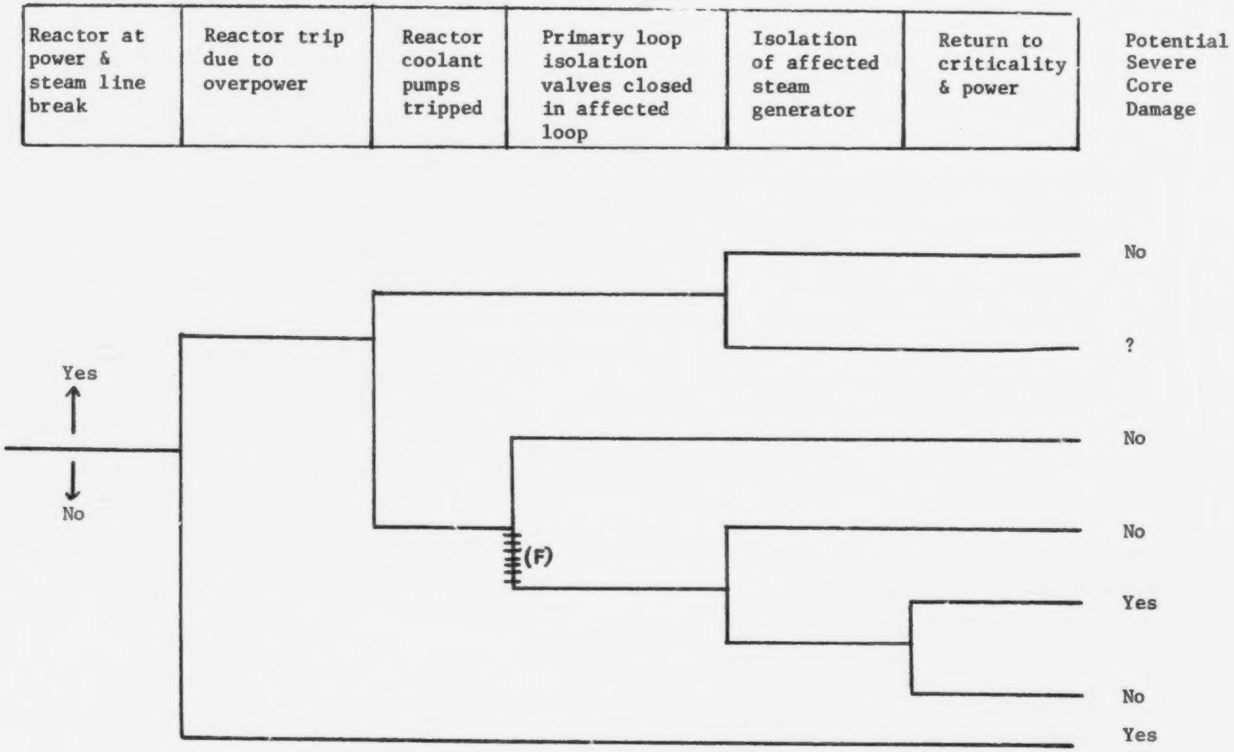
Unavailability of component per WASH 1400:* valves, motor operated, failure to operate: 10^{-3}/D.

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor in refueling shutdown and loop isolation valve testing in progress
Primary loop isolation valves operational
Potential Severe Core Damage
Yes
No
No
No — R_x shutdown and no steam line break

NSIC 32429 — Actual Occurrence for Loop Isolation Valve Failure Due to Improper Lubrication at Indian Point 1


Reactor at power & steam line break
Reactor trip due to overpower
Reactor coolant pumps tripped
Primary loop isolation valves closed in affected loop
Isolation of affected steam generator
Return to criticality & power
Potential Severe Core Damage
Yes
No
(F)
No
?
No
No
Yes
No
Yes

NSIC 32429 — Sequence of Interest for Loop Isolation Valve Failure at Indian Point 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 32429

DATE OF LER: March 13, 1969

SYSTEM INVOLVED: Primary Loop Isolation

COMPONENT INVOLVED: Valve operators

CAUSE: Improper lubrication (human error)

SEQUENCE OF INTEREST: Steam-line rupture.

ACTUAL OCCURRENCE: RC Loop Isolation Valves failed to close.

REACTOR NAME: Indian Point 1

DOCKET NUMBER: 50-3

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 265 MWe

REACTOR AGE: 7.5 yr

VENDOR: B&W

ARCHITECT-ENGINEERS: Con. Edison

OPERATORS: Con. Edison

LOCATION: 3 miles SW of Peekskill, NY

DURATION: 4380(a) hours

PLANT OPERATING CONDITION: Shut down

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; [✓ (b)] failed to start;
(c) made inoperable; (d) ________________

DISCOVERY METHOD: During test

COMMENT: If pumps fail to trip on steam line break, isolation valves prevent fast cooldown and return to criticality.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 36147

Date: July 15, 1969

Title: Loss of Offsite Power and Load Rejection at Haddam Neck

The failure sequence was:

1. One of the two 115KV offsite power lines (Montville line) was removed from service.
2. When the line was removed from service, trip inputs to the plant electrical system protection system were not defeated.
3. When the dispatcher opened other terminals on the Montville line, trip signals were generated which caused the two station service transformer low side breakers to open, resulting in a loss of offsite power.
4. Three of three diesel generators started and assumed safety related loads (one diesel generator loaded late).
5. One charging pump tripped during the starting sequence and one reactor coolant pump seal failed with excessive leakage (15 gpm seal injection required).

Corrective action:

1. Switching orders were revised to require defeat of trip signals from isolated lines.
2. The loop containing the leaking RCP seal was isolated for cooldown and evaluation.

Design purpose of failed system or component:

1. Offsite power provides electric power to safety-related components when the unit generator is inoperable.

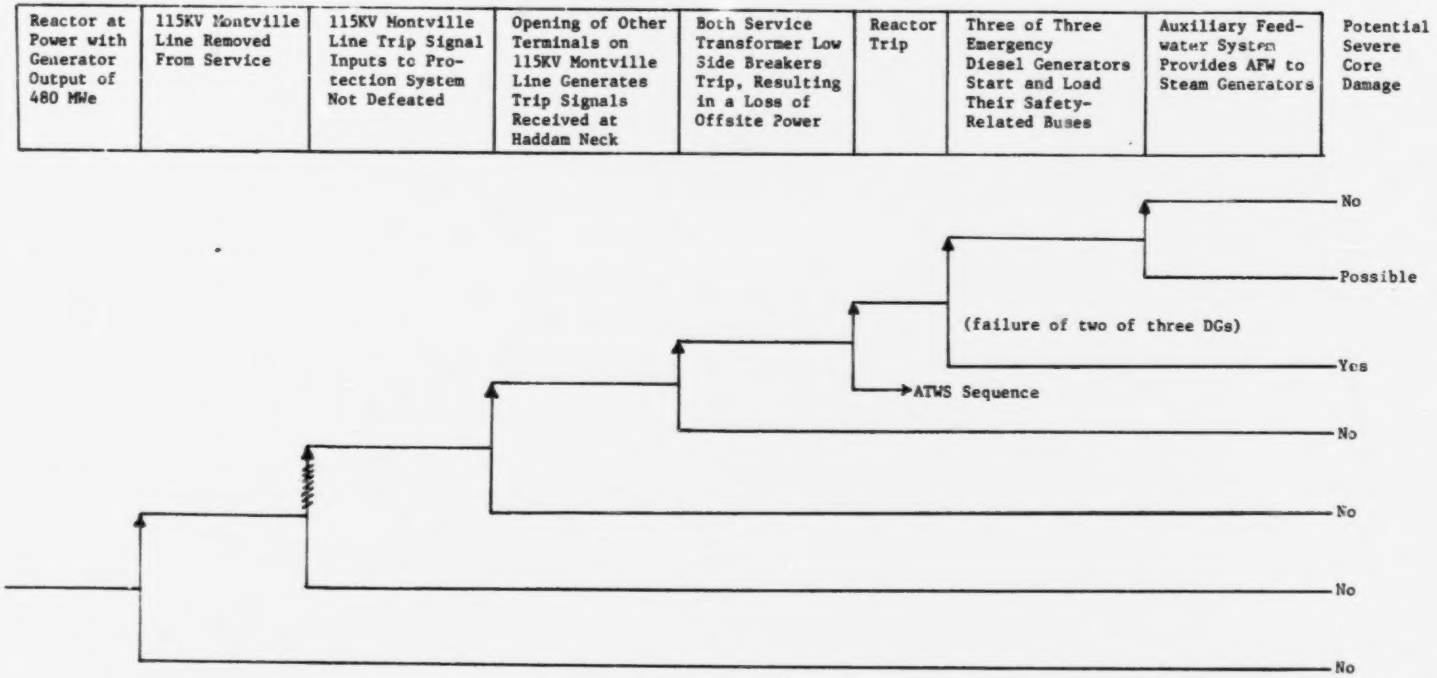
Unavailability of system per WASH 1400:* Offsite power: 2×10^{-5}/hr

Unavailability of component per WASH 1400:* —

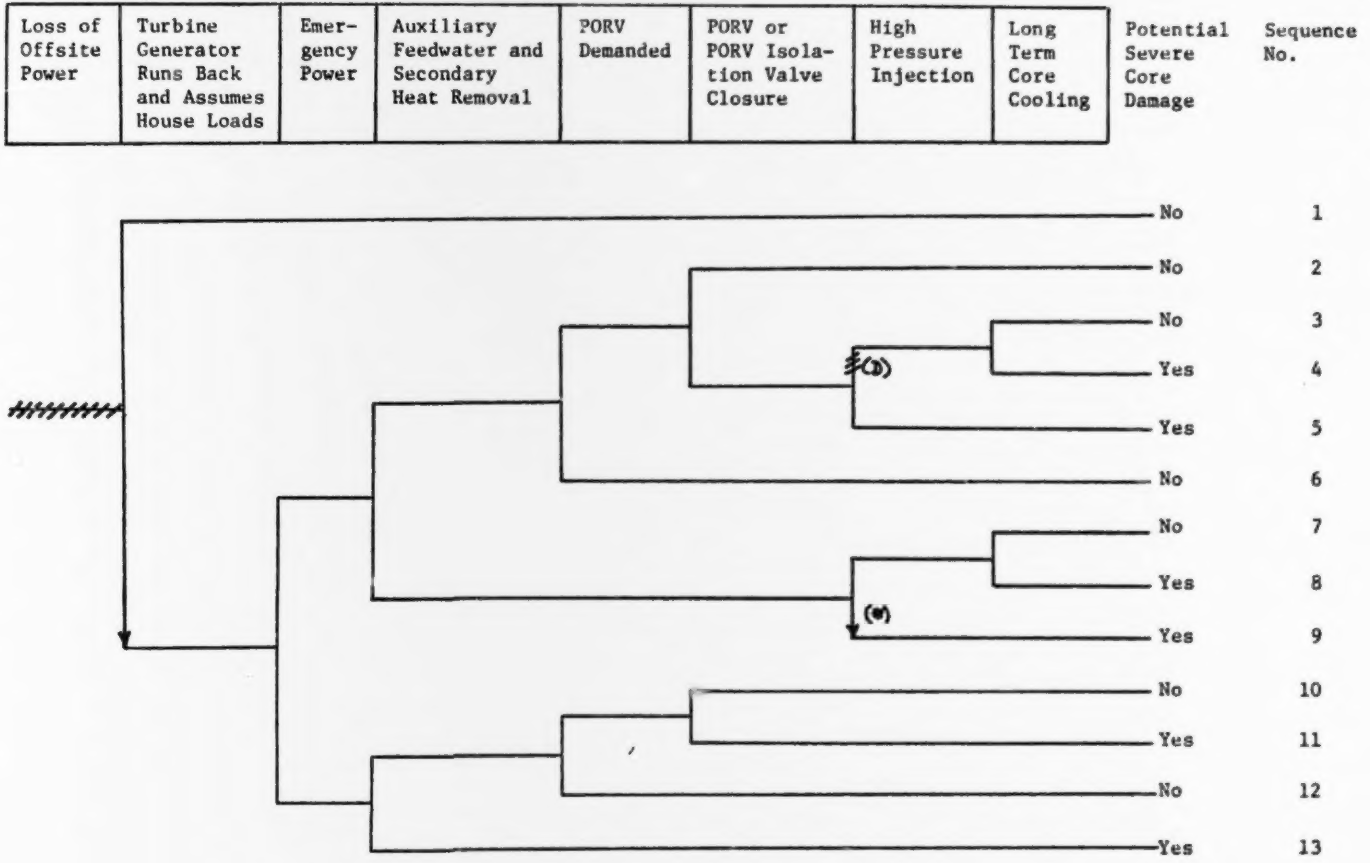
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

BEST DOCUMENT AVAILABLE


Reactor at Power with Generator Output of 480 MWe
115KV Montville Line Removed From Service
115KV Montville Line Trip Signal Inputs to Protection System Not Defeated
Opening of Other Terminals on 115KV Montville Line Generates Trip Signals Received at Haddam Neck
Both Service Transformer Low Side Breakers Trip, Resulting in a Loss of Offsite Power
Reactor Trip
Three of Three Emergency Diesel Generators Start and Load Their Safety-Related Buses
Auxiliary Feedwater System Provides AFW to Steam Generators
Potential Severe Core Damage
No
Possible
(failure of two of three DGs)
Yes
ATWS Sequence
No
No
No
No

NSIC 36147 — Actual Occurrence for Loss of Offsite Power and load Rejection at Haddam Neck


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
(D)
(*)
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
Yes 9
No 10
Yes 11
No 12
Yes 13

NSIC 36147 — Sequence of Interest for Loss of Offsite Power and load Rejection at Haddam Neck

*Use of HPI after failure of AFW was not included in mitigation procedure.

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: **36147**

DATE OF LER: **July 24, 1969**

DATE OF EVENT: **July 15, 1969**

SYSTEM INVOLVED: **electric power**

COMPONENT INVOLVED: **station service transformer low side breakers**

CAUSE: **Procedural error resulted in trip signals from an isolated line being received and interpreted as a trip signal related to the single operating offsite power line. Operator error.**

SEQUENCE OF INTEREST: **loss of offsite power**

ACTUAL OCCURRENCE: **reactor trip with loss of offsite power**

REACTOR NAME: **Haddam Neck**

DOCKET NUMBER: **50-213**

REACTOR TYPE: **PWR**

DESIGN ELECTRICAL RATING: **575** MWe

REACTOR AGE: **2.0** yr

VENDOR: **Westinghouse**

ARCHITECT-ENGINEERS: **Stone & Webster**

OPERATORS: **Connecticut Yankee Atomic Power Co.**

LOCATION: **13 miles east of Meridan, Conn.**

DURATION: **N/A**

PLANT OPERATING CONDITION: **at power**

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c) made inoperable]; (d) ____________________

DISCOVERY METHOD: **during operation**

COMMENT: –

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 39024

Date: January 25, 1972

Title: Loss of Off Site Power and Other Failures At Big Rock Point

The failure sequence was:

1. "Galloping Conductors" caused line faults which resulted in the failure of the Gaylord 388 oil circuit breaker (OCB).
2. Other breakers at the substation acted to clear the fault. This, however, left the plant in a no load condition, resulting in a turbine trip on overspeed and subsequently in a reactor trip due to high neutron flux.
3. The 199 OCB was manually opened since the 138 kv line was de-energized intermittently (unspecified reasons) for 20 minutes.
4. The station transferred to the 46 kv back-up line, however, the breaker tripped. A stuck contact of an instantaneous overcurrent relay (1288 OCB) in conjunction with operation of the undervoltage bus fault relay caused the 46 kv line to de-energize. (see attached page)

Corrective action:

1. An inspection of the transmission lines indicated they had received no damage during event.
2. The faulty trip coil (388 OCB) and the faulty overcurrent relay (1288 OCB) were repaired.

Design purpose of failed system or component:

1. Offsite electric power (both normal and backup) provide power to the station when the unit operator is not inservice.
2. Relays and circuit breakers are provided to protect electrical components from excessive and insufficient current and voltage conditions.

Unavailability of system per WASH 1400:* LOOP: 2×10^{-5}/Hr.

Unavailability of component per WASH 1400:*	circuit breakers	1×10^{-3}/D
	relays	1×10^{-4}/D

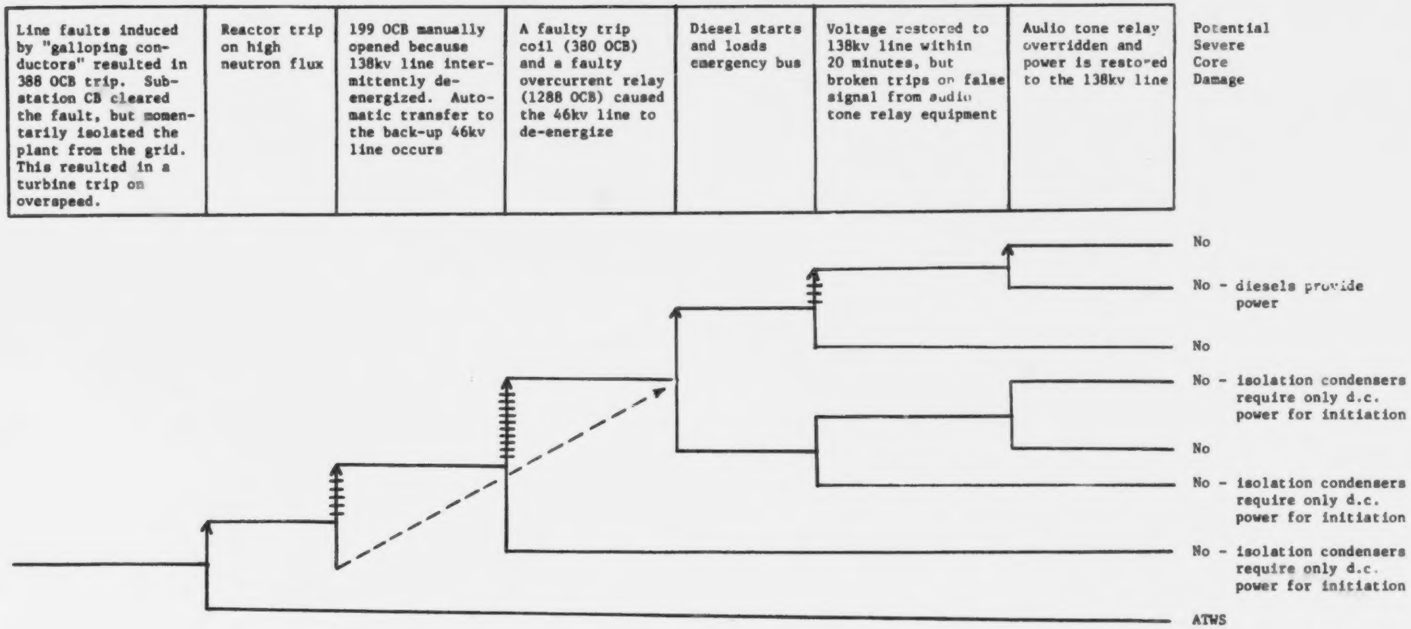
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

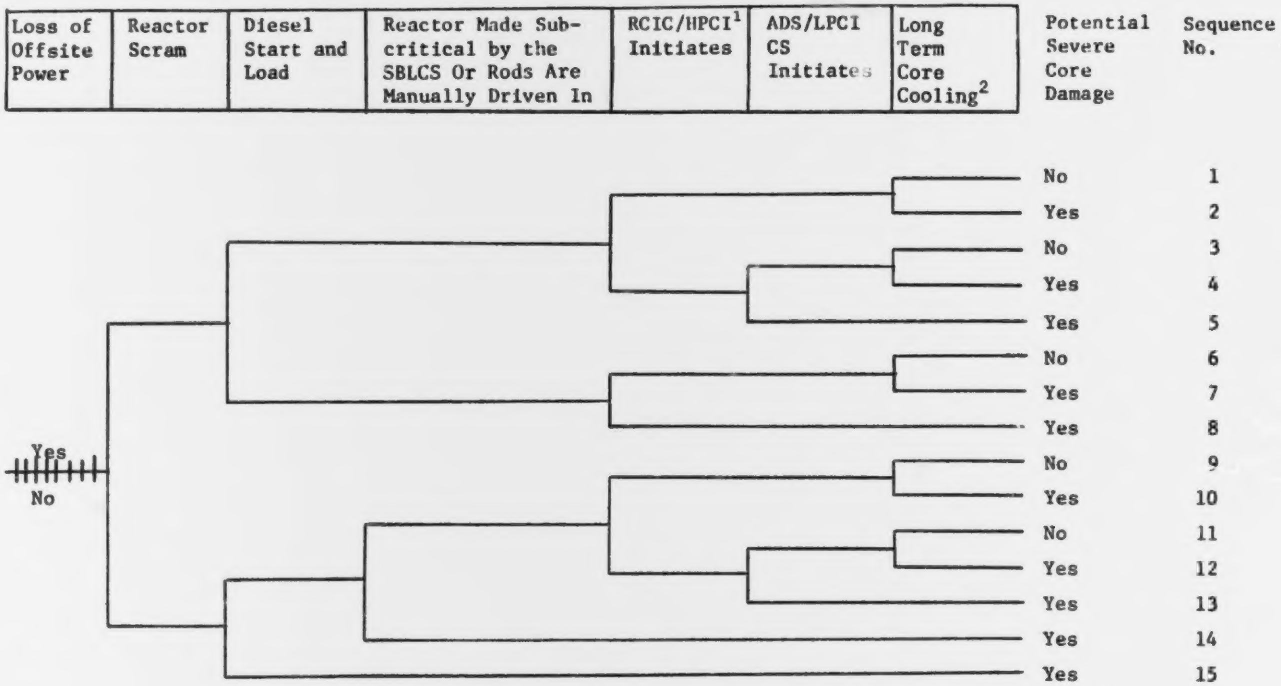
[redacted]

The failure sequence was: (continued)

5. Upon loss of both offsite lines the diesel generator started and loaded the 2-B bus.
6. The 138 kv line was re-energized approximately 20 minutes after the turbine trip. Attempts to close the 199 OCB failed due to false trip signals from the audio relay equipment.
7. The audio relay equipment was overridden and offsite power was restored.


Line faults induced by "galloping conductors" resulted in 388 OCB trip. Substation CB cleared the fault, but momentarily isolated the plant from the grid. This resulted in a turbine trip on overspeed.
Reactor trip on high neutron flux
199 OCB manually opened because 138kv line intermittently de-energized. Automatic transfer to the back-up 46kv line occurs
A faulty trip coil (380 OCB) and a faulty overcurrent relay (1288 OCB) caused the 46kv line to de-energize
Diesel starts and loads emergency bus
Voltage restored to 138kv line within 20 minutes, but broken trips on false signal from audio tone relay equipment
Audio tone relay overridden and power is restored to the 138kv line
Potential Severe Core Damage
No
No - diesels provide power
No
No - isolation condensers require only d.c. power for initiation
No
No - isolation condensers require only d.c. power for initiation
No - isolation condensers require only d.c. power for initiation
ATWS

NSIC 39024 — Actual Occurrence of Loss of Offsite Power and Other Failures at Big Rock Point


Loss of Offsite Power
Reactor Scram
Diesel Start and Load
Reactor Made Sub-critical by the SBLCS Or Rods Are Manually Driven In
RCIC/HPCI[1] Initiates
ADS/LPCI CS Initiates
Long Term Core Cooling[2]
Potential Severe Core Damage
Sequence No.
Yes
No
No 1
Yes 2
No 3
Yes 4
Yes 5
No 6
Yes 7
Yes 8
No 9
Yes 10
No 11
Yes 12
Yes 13
Yes 14
Yes 15

NSIC 39024 — Sequence of Interest of Loss of Offsite Power and Other Failures at Big Rock Point

[1] Big Rock Point utilizes isolation condenser instead of RCIC

[2] Success requires return of A.C. power within a day.

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: **39024**

DATE OF LER: **March 3, 1972**

DATE OF EVENT: **January 25, 1972**

SYSTEM INVOLVED: **electric power**

COMPONENT INVOLVED: **relays and circuit breakers**

CAUSE: **line faults induced by a violent storm**

SEQUENCE OF INTEREST: **loss of offsite power**

ACTUAL OCCURRENCE: **loss of offsite power**

REACTOR NAME: **Big Rock Point**

DOCKET NUMBER: **50-155**

REACTOR TYPE: **BWR**

DESIGN ELECTRICAL RATING: **72** MWe

REACTOR AGE: **9.3** yr

VENDOR: **General Electric**

ARCHITECT-ENGINEERS: **Bechtel**

OPERATORS: **Consumers Power Company**

LOCATION: **Four miles NE of Charlevoix, Mich.**

DURATION: **N/A**

PLANT OPERATING CONDITION: **just scrammed**

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________________

DISCOVERY METHOD: **operational event**

COMMENT:

24

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 39380

Date: December 8, 1971

Title: Safety Valve Operation After Feedwater Transient at Dresden 3

The failure sequence was:

1. Loss of the 3C condensate booster pump initiated a feedwater transient.
2. The 3B feedwater pump auto-started and was used to recover water level.
3. The 3B feedwater regulating valve locked out in the open position and the flow on light failed to indicate this situation.
4. Within 2 minutes and 45 seconds the water level had reached the main steamline causing a safety valve to actuate which resulted in pressurizing the drywell.
5. During the transient low power range monitor cabling, one electromatic relief valve solenoid operator and piping insulation were damaged.

Corrective action:

1. All damaged equipment was repaired or replaced.
2. The feedwater control system was recalibrated and the signals verified.
3. The air system was modified by installing a 10 ft^3 accumulator.

Design purpose of failed system or component:

The feedwater regulating valve controls the amount of feedwater supplied to the reactor.

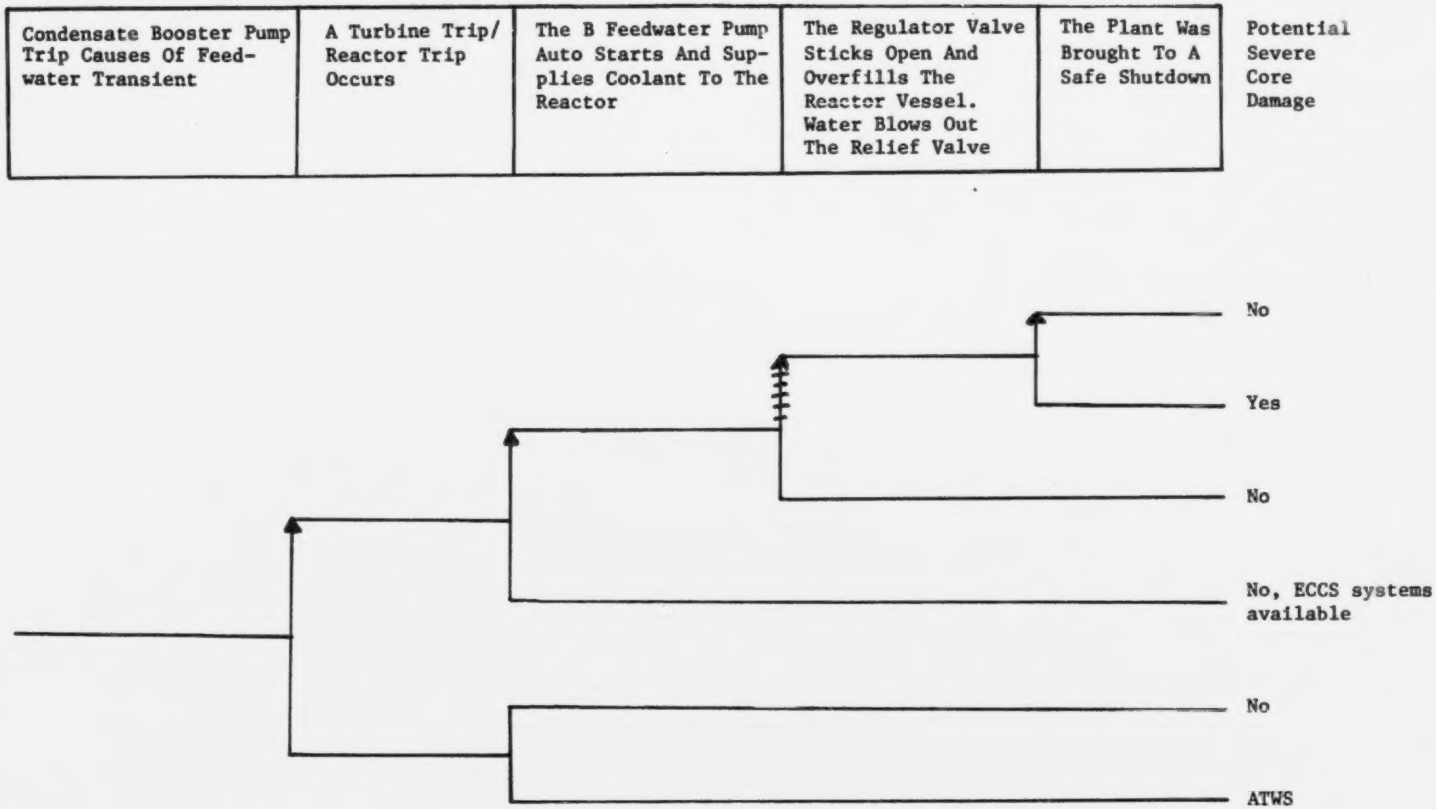
Unavailability of system per WASH 1400:* -

Unavailability of component per WASH 1400:* -

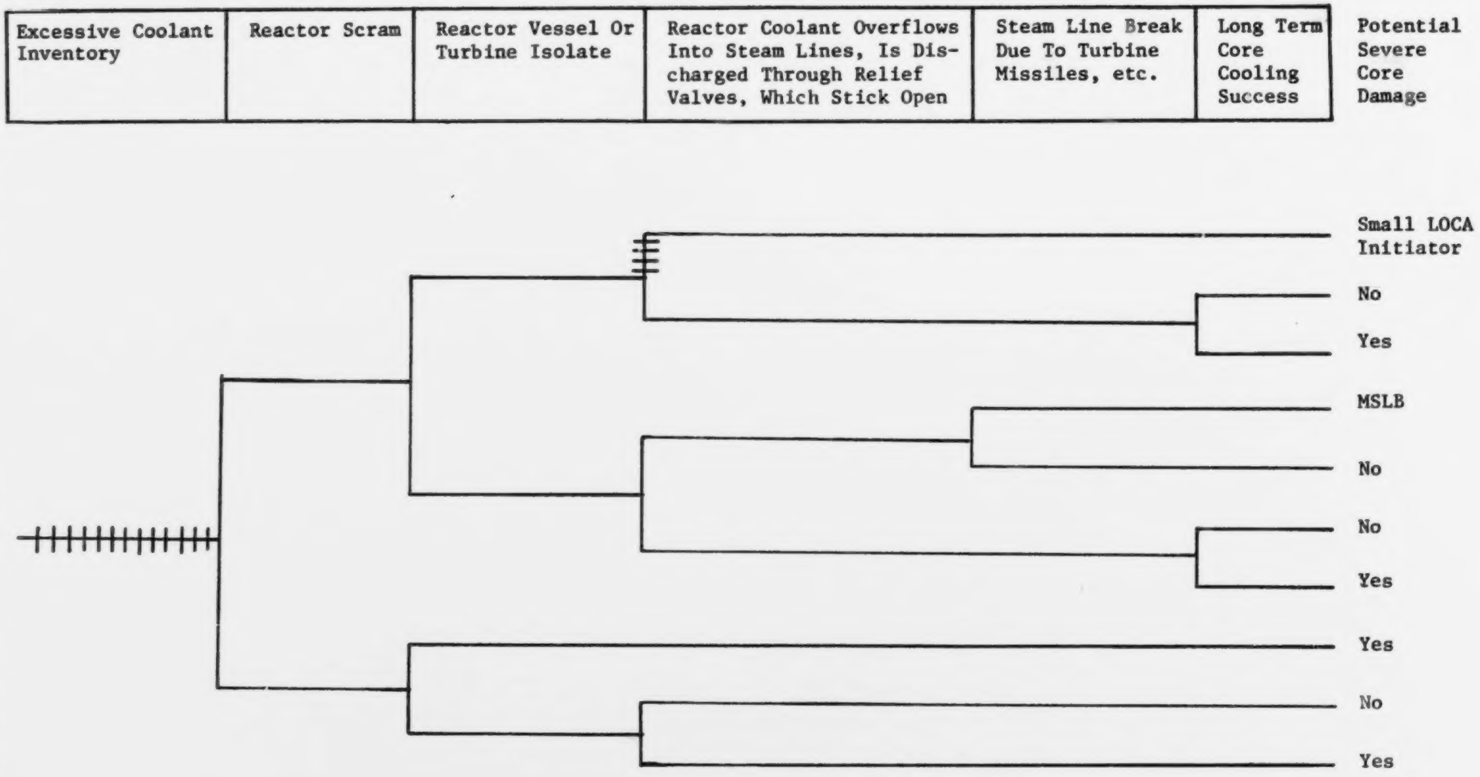
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

BEST DOCUMENT AVAILABLE


Condensate Booster Pump Trip Causes Of Feedwater Transient
A Turbine Trip/ Reactor Trip Occurs
The B Feedwater Pump Auto Starts And Supplies Coolant To The Reactor
The Regulator Valve Sticks Open And Overfills The Reactor Vessel. Water Blows Out The Relief Valve
The Plant Was Brought To A Safe Shutdown
Potential Severe Core Damage
No
Yes
No
No, ECCS systems available
No
ATWS

NSIC 39380 — Actual Occurrence of Safety Valve Actuation After Feedwater Transient at Dresden 3


Excessive Coolant Inventory
Reactor Scram
Reactor Vessel Or Turbine Isolate
Reactor Coolant Overflows Into Steam Lines, Is Discharged Through Relief Valves, Which Stick Open
Steam Line Break Due To Turbine Missiles, etc.
Long Term Core Cooling Success
Potential Severe Core Damage
Small LOCA Initiator
No
Yes
MSLB
No
No
Yes
Yes
No
Yes

NSIC 39380 — Sequence of Interest for Safety Valve Actuation After Feedwater Transient at Dresden 3

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 39380

DATE OF LER: February 22, 1972

DATE OF EVENT: December 8, 1971

SYSTEM INVOLVED: Pressure Relief, Feedwater

COMPONENT INVOLVED: Feedwater Regulator Valve

CAUSE: A mechanical failure

SEQUENCE OF INTEREST: Excessive Coolant Inventory Transient

ACTUAL OCCURRENCE: Safety valve operation after feedwater transient

REACTOR NAME: Dresden 3

DOCKET NUMBER: 50-249

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 794 MWe

REACTOR AGE: 0.94 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: Sargent & Lundy

OPERATORS: Commonwealth Edison

LOCATION: Nine miles East of Morris, Ill.

DURATION: N/A

PLANT OPERATING CONDITION: 100% power

SAFETY FEATURE TYPE OF FAILURE: [✓ (a)] inadequate performance; (b) failed to start; (c) made inoperable; (d) ____________

DISCOVERY METHOD: operational transient

COMMENT:

BEST DOCUMENT AVAILABLE

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 44751

Date: April 27, 1970

Title: **Failure of Three of Four Safety System Sensors for Primary Drum Low Level Scram at Dresden 1**

The failure sequence was:

During a routine test, three of four primary drum low level scram system sensors were found failed due to crud build-up in the switches. These failures would cause a failure to scram on low drum level if that were required.

Corrective action:

The switches were flushed and their proper operation was demonstrated. The switches were to be replaced with differential-type switches at the up-coming refueling.

Design purpose of failed system or component:

The switches provide trip input to the scram system for primary drum low level.

Unavailability of system per WASH 1400:* --

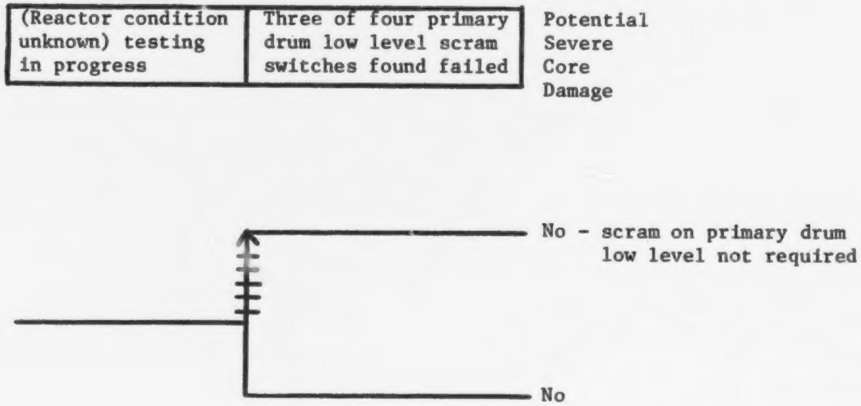
Unavailability of component per WASH 1400:* float switches not considered in WASH-1400.

---

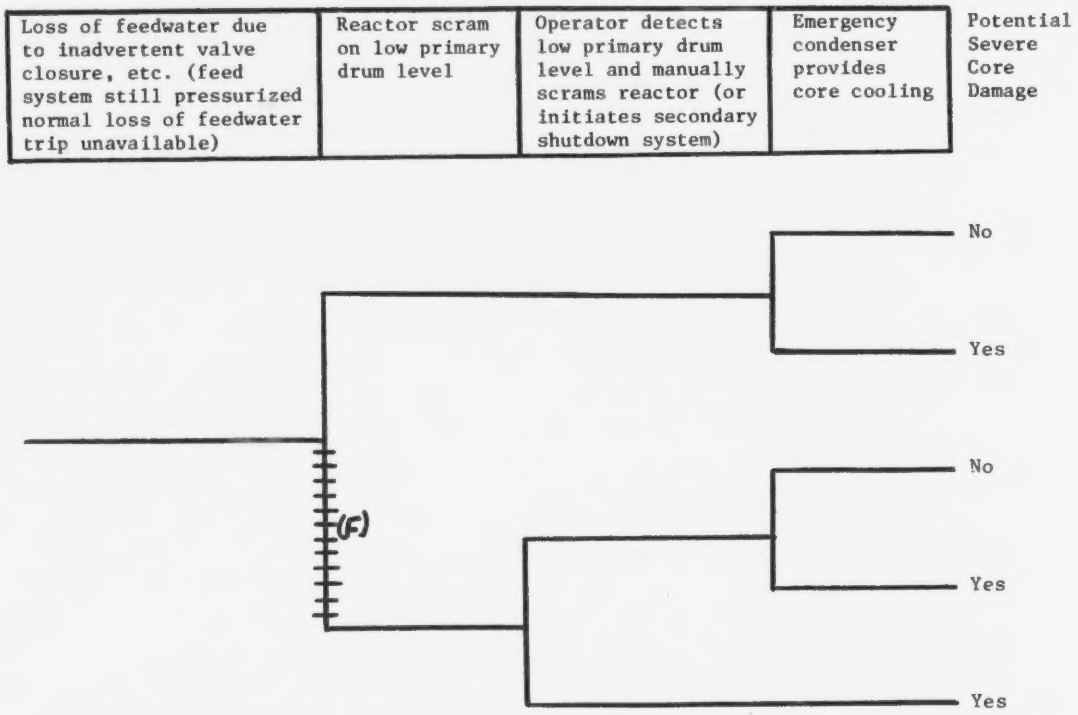
*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

Blank

Page


(Reactor condition unknown) testing in progress
Three of four primary drum low level scram switches found failed
Potential Severe Core Damage
No - scram on primary drum low level not required
No

NSIC 44751 — Actual Occurrence for Failure of Three of Four Safety System Sensors for Primary Drum Level Scram at Dresden 1


Loss of feedwater due to inadvertent valve closure, etc. (feed system still pressurized normal loss of feedwater trip unavailable)
Reactor scram on low primary drum level
Operator detects low primary drum level and manually scrams reactor (or initiates secondary shutdown system)
Emergency condenser provides core cooling
Potential Severe Core Damage
No
Yes
No
Yes
Yes
(F)

NSIC 44751 — Sequence of Interest for Failure of Three of Four Safety System Sensors for Primary Drum Level Scram at Dresden 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 44751

DATE OF LER: April 27, 1970

DATE OF EVENT: unknown

SYSTEM INVOLVED: Reactor Protection System

COMPONENT INVOLVED: Primary Drum Level Float Switches

CAUSE: Crud buildup in the float chambers

SEQUENCE OF INTEREST: Loss of Main Feedwater

ACTUAL OCCURRENCE: Switch failure during testing

REACTOR NAME: Dresden 1

DOCKET NUMBER: 50-10

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 200 MWe

REACTOR AGE: 10.5 yr

VENDOR: GE

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Commonwealth Edison Co.

LOCATION: Nine miles E of Morris, Ill.

DURATION: 360 (a) hours

PLANT OPERATING CONDITION: unknown

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________________

DISCOVERY METHOD: during testing

COMMENT: only abstract available

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 47814

Date: June 5, 1970

Title: Depressurization Incident at Dresden 2

The failure sequence was:

1. A spurious signal in the reactor pressure control system occurred causing the turbine control valve to open from 75% to 80% and turbine bypass valve to open 100%.
2. The reactor scrammed and the water level began to drop. The pen in the water level stuck on the low water level mark. Due to this incorrect signal the operator took manual control of the feedwater system and began to increase flow to the reactor.
3. After no change in water level indication the operator realized the pen was stuck. He tapped the pen, however water was already blowing through the relief valves.
4. The feedwater flow was throttled back.

Corrective action:

The plant was shutdown for an extended period for repair of minor damage and inspection.

Design purpose of failed system or component:

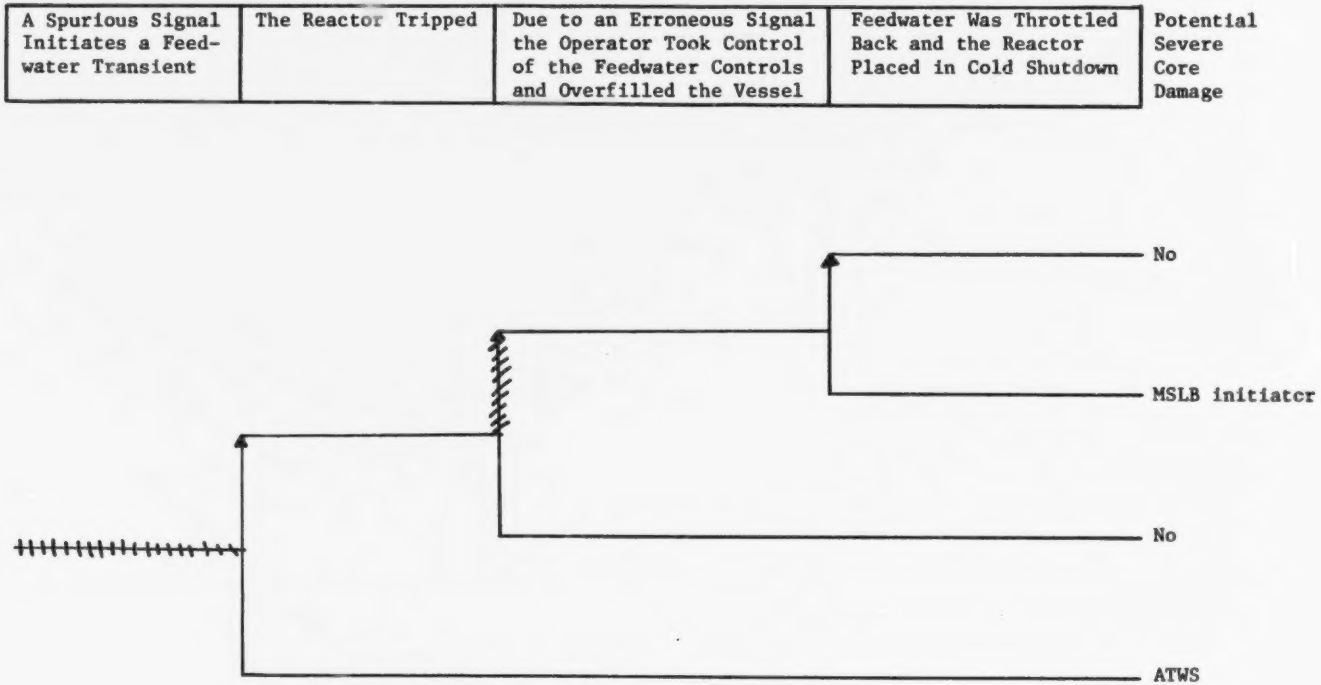
The reactor water level instrumentation indicates the water level in the reactor vessel.

Unavailability of system per WASH 1400:*

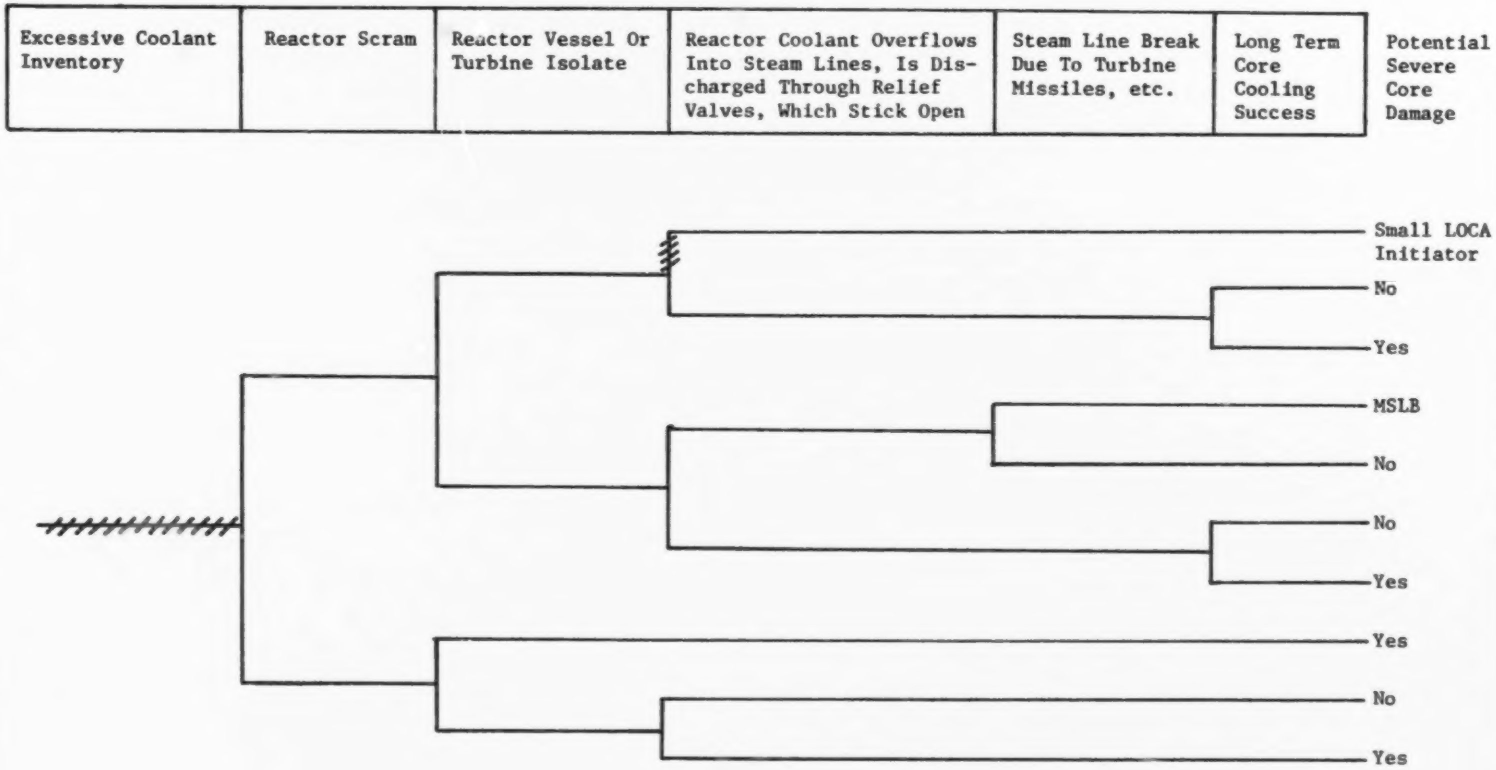
Unavailability of component per WASH 1400:*

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


A Spurious Signal Initiates a Feed-water Transient
The Reactor Tripped
Due to an Erroneous Signal the Operator Took Control of the Feedwater Controls and Overfilled the Vessel
Feedwater Was Throttled Back and the Reactor Placed in Cold Shutdown
Potential Severe Core Damage
No
MSLB initiatcr
No
ATWS

NSIC 47814 — Actual Occurrence of the Depressurization Incident at Dresden 2


Excessive Coolant Inventory
Reactor Scram
Reactor Vessel Or Turbine Isolate
Reactor Coolant Overflows Into Steam Lines, Is Discharged Through Relief Valves, Which Stick Open
Steam Line Break Due To Turbine Missiles, etc.
Long Term Core Cooling Success
Potential Severe Core Damage
Small LOCA Initiator
No
Yes
MSLB
No
No
Yes
Yes
No
Yes

NSIC 47814 — Sequence of Interest for the Depressurization Incident at Dresden 2

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 47814

DATE OF LER: July 6, 1970

DATE OF EVENT: June 5, 1970

SYSTEM INVOLVED: Feedwater

COMPONENT INVOLVED: Reactor Water Level Indicator.

CAUSE: Component failure

SEQUENCE OF INTEREST: Excessive Coolant Inventory

ACTUAL OCCURRENCE: Depressurization Incident

REACTOR NAME: Dresden 2

DOCKET NUMBER: 50-237

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 794 MWe

REACTOR AGE: 0.4 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: Sargents and Lundy

OPERATORS: Commonwealth Edison

LOCATION: Nine miles from Morris, Ill.

DURATION: N/A

PLANT OPERATING CONDITION: 75%

SAFETY FEATURE TYPE OF FAILURE: [✓ (a)] inadequate performance; (b) failed to start; (c) made inoperable; (d) ______________________

DISCOVERY METHOD: operational event

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 59484

Date: January 22, 1971

Title: Failure of Containment Sump Isolation Valves at Point Beach 1

The failure sequence was:

1. During a routine check of the containment tendon access gallery with the plant at power, air was observed leaking from the packing of one of the sump isolation valves.
2. When an attempt was made to open the valve it would not open due to a shorted solenoid in the valve hydraulic positioner.
3. The redundant containment sump isolation valve was also found inoperable because of a stuck solenoid in the valve hydraulic positioner.

Corrective action:

1. Both valves were repaired and tested.
2. Both valves were added to the list of valves requiring a monthly check to verify operability.

Design purpose of failed system or component:

The valves isolate the containment sump from the low pressure injection/recirculation system during normal operation. After the RWST is emptied following a LOCA, these valves are opened to provide suction flow to the low pressure recirculation system.

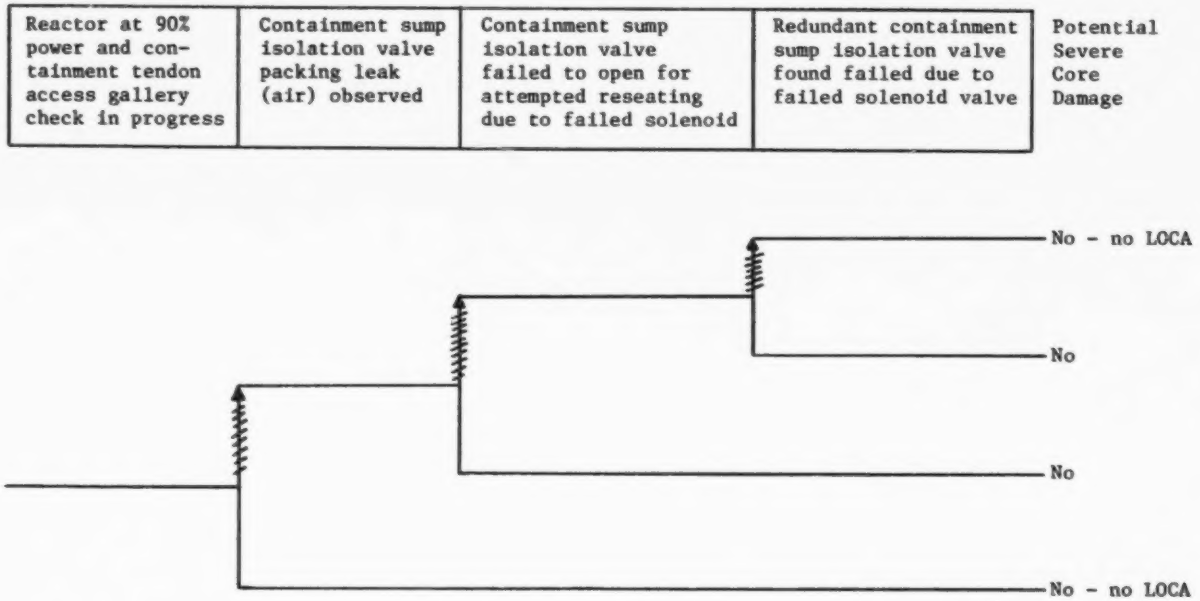
Unavailability of system per WASH 1400:[*] low pressure recirculation: 1.3×10^{-2}/D

Unavailability of component per WASH 1400:[*] valves, motor operated, failure to operate: 1×10^{-3}/D

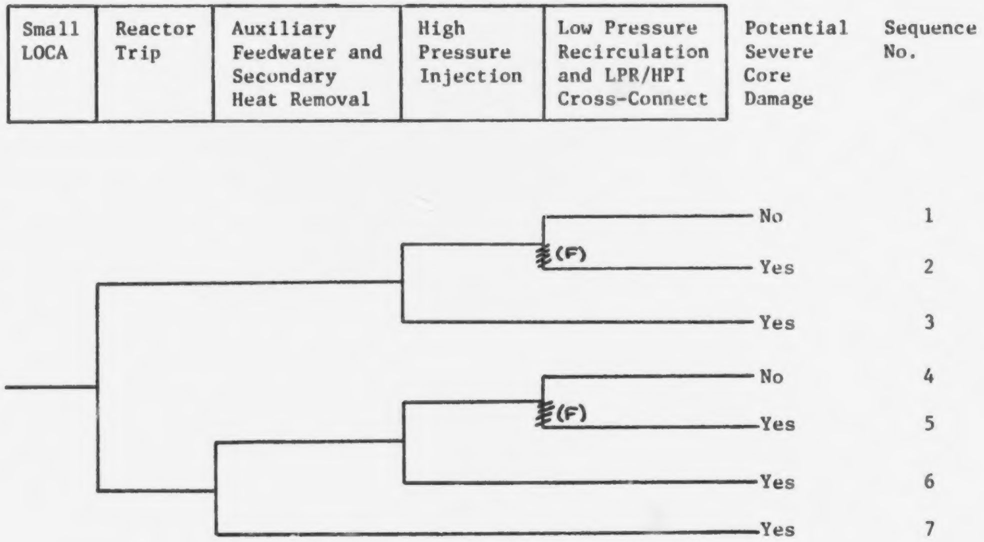
valves, air-fluid operated, failure to operate: 3×10^{-4}/D

---

[*] Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor at 90% power and containment tendon access gallery check in progress
Containment sump isolation valve packing leak (air) observed
Containment sump isolation valve failed to open for attempted reseating due to failed solenoid
Redundant containment sump isolation valve found failed due to failed solenoid valve
Potential Severe Core Damage
No - no LOCA
No
No
No - no LOCA

NSIC 59484 — Actual Occurrence for Failure of Sump Isolation Valves at Point Beach 1


Small LOCA
Reactor Trip
Auxiliary Feedwater and Secondary Heat Removal
High Pressure Injection
Low Pressure Recirculation and LPR/HPI Cross-Connect
Potential Severe Core Damage
Sequence No.
(F)
(F)
No 1
Yes 2
Yes 3
No 4
Yes 5
Yes 6
Yes 7

NSIC 59484 — Sequence of Interest for Failure of Sump Isolation Valves at Point Beach 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 59484

DATE OF LER: January 22, 1971

DATE OF EVENT: January 12, 1971

SYSTEM INVOLVED: low pressure recirculation

COMPONENT INVOLVED: containment sump isolation valves

CAUSE: valves failed closed due to failed solenoids in the hydraulic operators

SEQUENCE OF INTEREST: LOCA

ACTUAL OCCURRENCE: sump isolation valves failed closed during testing

REACTOR NAME: Point Beach 1

DOCKET NUMBER: 50-266

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 497 MWe

REACTOR AGE: .2 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Wisconsin Electric Power Co.

LOCATION: 15 miles north of Manitowoc, Wisc.

DURATION: 1750 hours

PLANT OPERATING CONDITION: 90% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start;
[✓ (c)] made inoperable; (d) ______________________

DISCOVERY METHOD: area surveillance

COMMENT:

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: **60227**

Date: **December 4, 1970**

Title: **Failure of Several MSIVs to Close at Dresden 2**

The failure sequence was:

1. **The plant was operating 299 MWe and the MSIVs were undergoing surveillance testing.**
2. **MSIVs 1D, 2A, 2B, 2C failed to close due to a thin material film on the valve pilot valves.**
3. **MSIV 1A, 2D failed the timing test due to the same cause.**
4. **Reactor shutdown as required by tech specs.**

Corrective action:

1. **The valves were cleaned.**
2. **The air supply was blown down to remove any particulates.**

Design purpose of failed system or component:

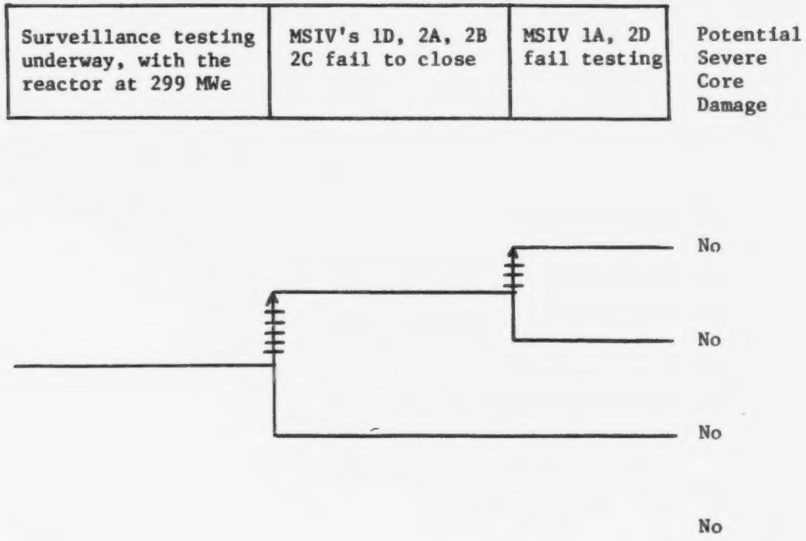
The MSIVs are designed to isolate the reactor vessel during transient conditions.

Unavailability of system per WASH 1400:* **-**

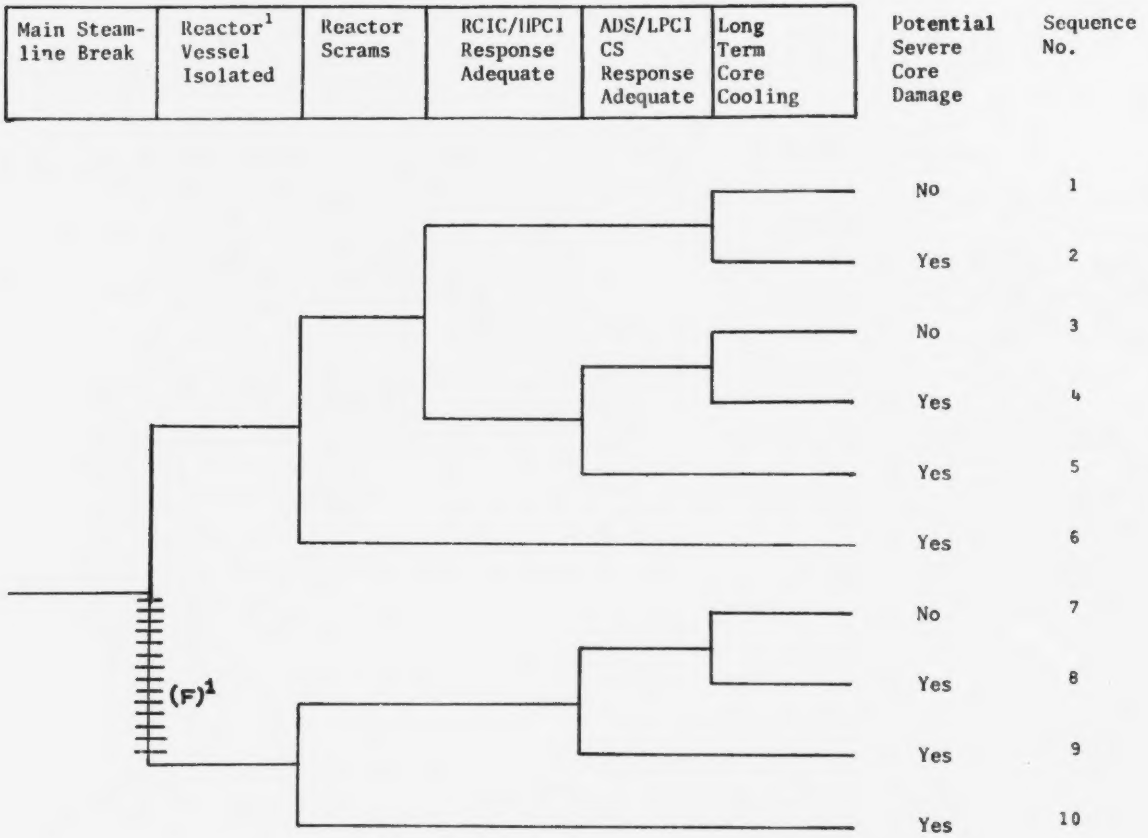
Unavailability of component per WASH 1400:* **air operated valve 3×10^{-4}/D**

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Surveillance testing underway, with the reactor at 299 MWe
MSIV's 1D, 2A, 2B 2C fail to close
MSIV 1A, 2D fail testing
Potential Severe Core Damage
No
No
No
No

NSIC 60227 — Actual Occurrence of Failure of Several MSIV s to Close at Dresden 2


Main Steam-line Break
Reactor[1] Vessel Isolated
Reactor Scrams
RCIC/HPCI Response Adequate
ADS/LPCI CS Response Adequate
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
Yes 2
No 3
Yes 4
Yes 5
Yes 6
No 7
Yes 8
Yes 9
Yes 10
(F)1

NSIC 60227 — Sequence of Interest for Failure of Several MSIV's at Dresden 2

[1] A failed state would result if the steamline break was between the two MSIV's on the D steam line. The failed MSIVs could just as easily occurred on any steam line.

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: **60227**

DATE OF LER: **December 11, 1970**

DATE OF EVENT: **December 4, 1970**

SYSTEM INVOLVED: **MSIV's**

COMPONENT INVOLVED: **MSIV's**

CAUSE: **dirty pilot operator valves**

SEQUENCE OF INTEREST: **Main steam line break**

ACTUAL OCCURRENCE: **failure of several MSIV's to close**

REACTOR NAME: **Dresden 2**

DOCKET NUMBER: **50-237**

REACTOR TYPE: **BWR**

DESIGN ELECTRICAL RATING: **794** MWe

REACTOR AGE: **.9** yr

VENDOR: **GE**

ARCHITECT-ENGINEERS: **Sargent and Lundy**

OPERATORS: **Commonwealth Edison**

LOCATION: **Nine miles E of Morris, Illinois**

DURATION: **4380 hours**

PLANT OPERATING CONDITION: **299 MWe (37.7% power)**

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; (c) made inoperable; [✓ (d)] **failed to close**

DISCOVERY METHOD: **surveillance testing**

COMMENT: **-**

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 61043

Date: February 10, 1971

Title: Plant Electrical Power Failure at LaCrosse

The failure sequence was:

1. With the reactor shutdown, maintenance was being performed at the Genoa substation.
2. Vibration resulting from that maintenance work tripped a protective differential relay, which resulted in a loss of power to the plant 69 KV tie line.
3. The diesel started and assumed safety-related loads.

Corrective action:

1. The emergency shutdown condenser and emergency core spray pump were used for core cooling.
2. Power was restored to the 69 KV tie line 14 minutes later.
3. Additional procedures to minimize inadvertant relay actuation during maintenance were initiated.

Design purpose of failed system or component:

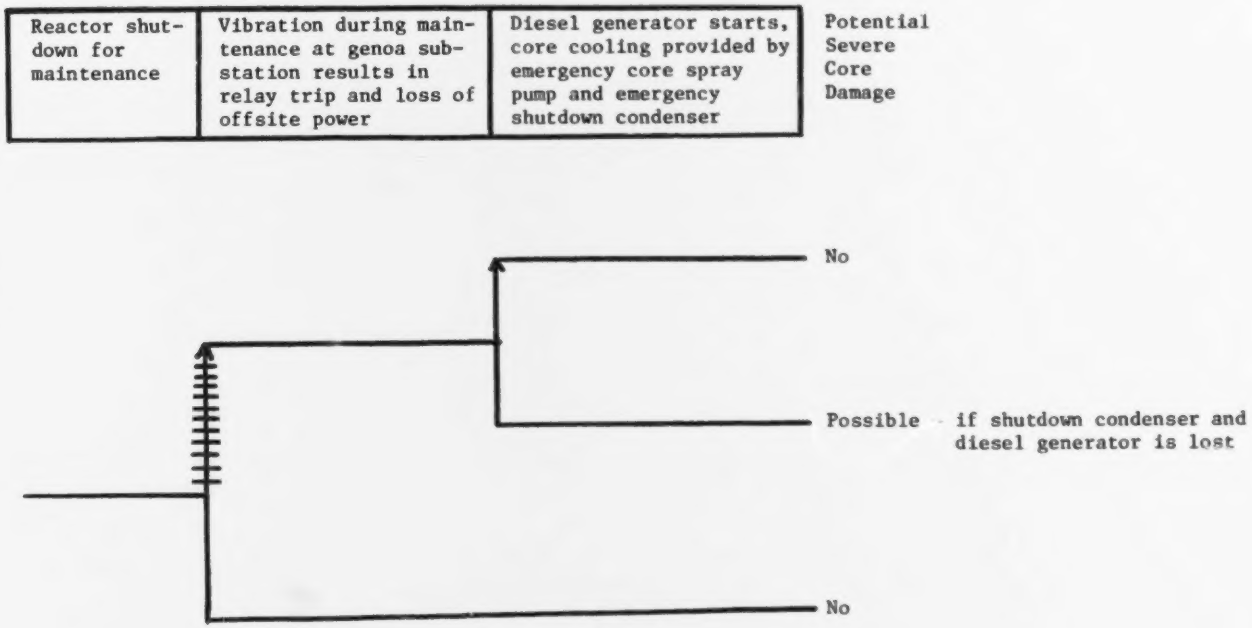
1. Offsite power provides power to plant safety-related loads when the unit generator is unavailable.

Unavailability of system per WASH 1400:* LOOP: 2×10^{-5}/hr.

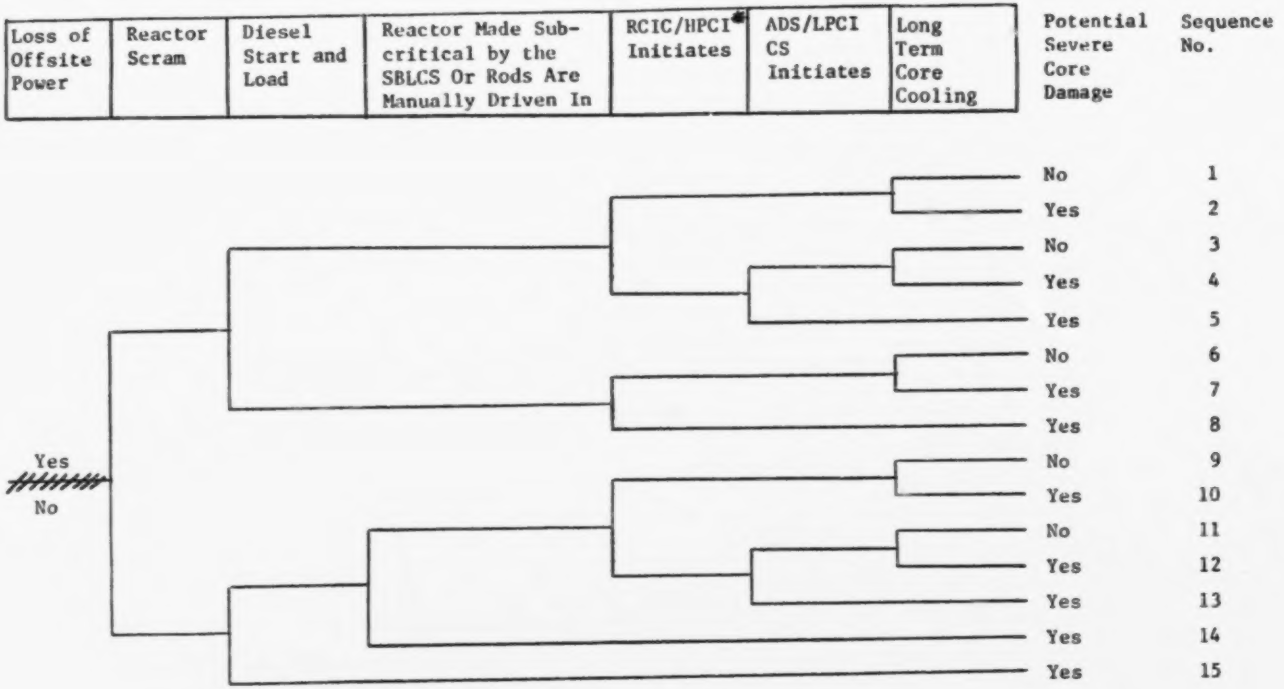
Unavailability of component per WASH 1400:* --

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor shutdown for maintenance
Vibration during maintenance at genoa substation results in relay trip and loss of offsite power
Diesel generator starts, core cooling provided by emergency core spray pump and emergency shutdown condenser
Potential Severe Core Damage
No
Possible - if shutdown condenser and diesel generator is lost
No

NSIC 61043 – Actual Occurrence for Plant Electrical Power Failure at La Crosse


Loss of Offsite Power
Reactor Scram
Diesel Start and Load
Reactor Made Sub-critical by the SBLCS Or Rods Are Manually Driven In
RCIC/HPCI*
Initiates
ADS/LPCI CS Initiates
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
No 1
Yes 2
No 3
Yes 4
Yes 5
No 6
Yes 7
Yes 8
No 9
Yes 10
No 11
Yes 12
Yes 13
Yes 14
Yes 15

NSIC 61043 — Sequence of Interset for Plant Electrical Power Failure at La Crosse

*La Crosse utilizes shutdown condenser and condensate pumps instead of RCIC and HPCI.

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 61043

DATE OF LER: February 10, 1971

DATE OF EVENT: January 20, 1971

SYSTEM INVOLVED: offsite power

COMPONENT INVOLVED: relay

CAUSE: inadvertent actuation during maintenance (human error)

SEQUENCE OF INTEREST: loss of offsite power

ACTUAL OCCURRENCE: loss of offsite power during shutdown

REACTOR NAME: LaCrosse

DOCKET NUMBER: 50-409

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 50 MWe

REACTOR AGE: 3.5 yr

VENDOR: Allis-Chalmers

ARCHITECT-ENGINEERS: Sargent & Lundy

OPERATORS: Dairyland Power Cooperative

LOCATION: 19 miles S of LaCrosse, Wisc.

DURATION: N/A

PLANT OPERATING CONDITION: shutdown

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start;
[✓ (c)] made inoperable; (d) ____________________

DISCOVERY METHOD: operational event

COMMENT: --

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 61434

Date: March 4, 1971

Title: Loss of Offsite Power and Failure of an Emergency Condensor Valve to Open at Humboldt Bay (Semi-Annual Report July-December 1970)

The failure sequence was:
1. With the reactor at power, a switching error at the Humboldt Substation caused protective relaying which resulted in a generator and turbine trip, loss of the 60 KV bus, and consequent loss of offsite power.
2. Loss of normal power resulted in reactor scram, loss of feedwater flow, loss of drywell cooling, and loss of control room indication of reactor vessel pressure and level.
3. The emergency propane generator started and assumed safety-related loads. (A back-up feed was available from the adjacent fossil unit.) A CRD hydraulic pump was started to provide reactor makeup.
4. The emergency condenser return valve failed closed due to an incorrectly adjusted torque switch.

cont.

Corrective action:
1. Normal power was restored to the plant 2.4 KV bus from an adjacent fossil unit.
2. The emergency condenser isolation valve was repaired and returned to service.

Design purpose of failed system or component:
1. Offsite power provides power for safety-related systems when the unit generator is unavailable.
2. The emergency condenser provides a means of removing reactor decay heat when the condenser is not available for steam dump.

Unavailability of system per WASH 1400:* LOOP: 2×10^{-5}/hr.

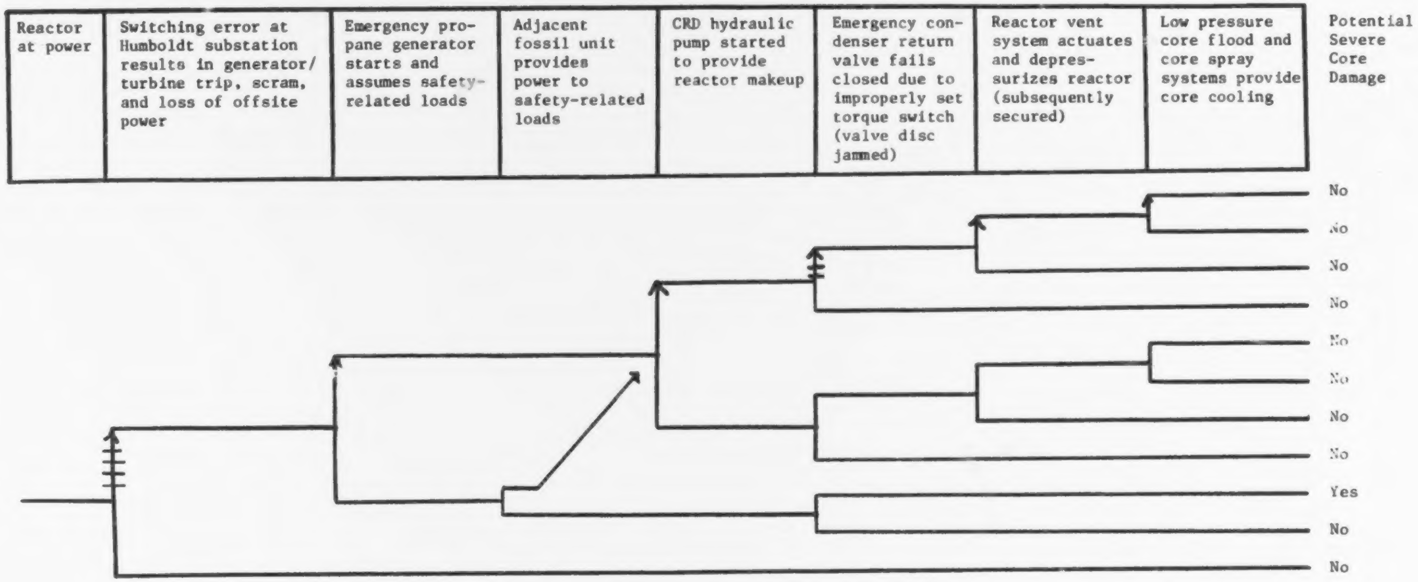
Unavailability of component per WASH 1400:* valve, motor operated, failure to open: 1×10^{-3}/D failure sequence continued

---

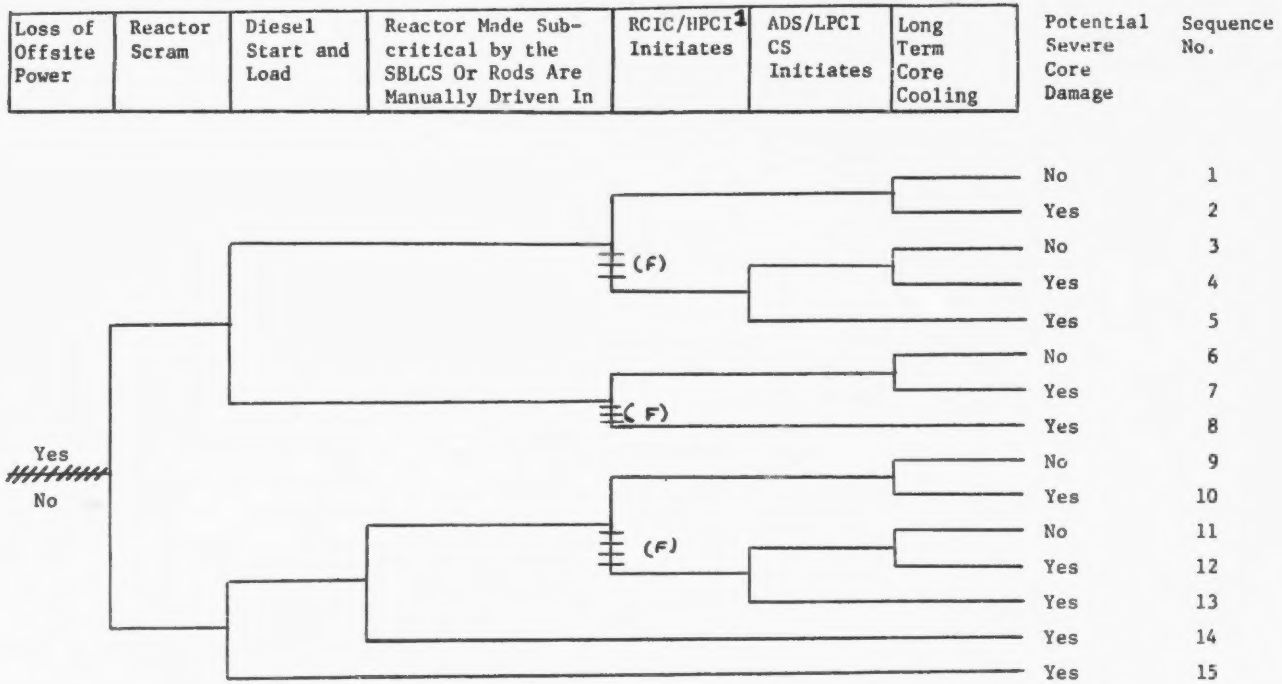
*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

5. Reactor vessel level shrink to the low water level setpoint due to the opening of a safety valve (a result of the failed emergency condenser) combined with the loss of feedwater and an increase in drywell pressure (due to loss of drywell cooling plus a failed relief valve expansion joint) resulted in the actuation of the reactor vent system. This system was manually secured four minutes later.

6. The low pressure core flood and core spray systems subsequently automatically initiated and were used for core cooling until normal power was restored.


Reactor at power
Switching error at Humboldt substation results in generator/turbine trip, scram, and loss of offsite power
Emergency propane generator starts and assumes safety-related loads
Adjacent fossil unit provides power to safety-related loads
CRD hydraulic pump started to provide reactor makeup
Emergency condenser return valve fails closed due to improperly set torque switch (valve disc jammed)
Reactor vent system actuates and depressurizes reactor (subsequently secured)
Low pressure core flood and core spray systems provide core cooling
Potential Severe Core Damage
No
No
No
No
No
No
No
No
Yes
No
No

NSIC 61434 — Actual Occurrence for Loss of Offsite Power and Failure of an Emergency Condenser Valve to Open at Humboldt Bay


Loss of Offsite Power
Reactor Scram
Diesel Start and Load
Reactor Made Subcritical by the SBLCS Or Rods Are Manually Driven In
RCIC/HPCI[1] Initiates
ADS/LPCI CS Initiates
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
(F)
(F)
(F)
No 1
Yes 2
No 3
Yes 4
Yes 5
No 6
Yes 7
Yes 8
No 9
Yes 10
No 11
Yes 12
Yes 13
Yes 14
Yes 15

NSIC 61434 — Sequence of Interest for Loss of Offsite Power and Failure of an Emergency Condenser Valve to Open at Humboldt Bay

[1] Humboldt Bay utilized an emergency condenser and CRD hydraulic pumps/safety valves for decay heat removal

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 61434

DATE OF LER: March 4, 1971

DATE OF EVENT: July 17, 1970

SYSTEM INVOLVED: offsite power, emergency condenser

COMPONENT INVOLVED: 60 KV bus, motor-operated valve

CAUSE: power was lost to the bus due to a switching error, the valve failed to operate due to an improperly set torque switch, operator error.

SEQUENCE OF INTEREST: loss of offsite power

ACTUAL OCCURRENCE: Loss of offsite power

REACTOR NAME: Humboldt Bay

DOCKET NUMBER: 50-133

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 65 MWe

REACTOR AGE: 7.4 yr

VENDOR: GE

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Pacific Gas and Electric Co.

LOCATION: 4 miles S of Eureka

DURATION: N/A

PLANT OPERATING CONDITION: at power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; (c) made inoperable; [✓ (d)] failed to open

DISCOVERY METHOD: operational event

COMMENT: —

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: **61565**

Date: **September 2, 1971**

Title: **Loss of Offsite Power and Failure of a Diesel Generator to Load at Palisades**

The failure sequence was:

1. **The Argenta No. 2 345KV line (one of three 345KV lines at the switchyard) tripped.**
2. **Failure of a "breaker failure relay" associated with the tripped line breaker resulted in the tripping of two other breakers on the ring bus, which caused a loss of offsite power.**
3. **Diesel generator No. 1 started and loaded its safety-related bus.**
4. **Diesel generator No. 2 started but its breaker did not close until the operator adjusted the synchroscope in preparation to close the breaker manually.**

Corrective action:

Not specified (only LER abstract available).

Design purpose of failed system or component:

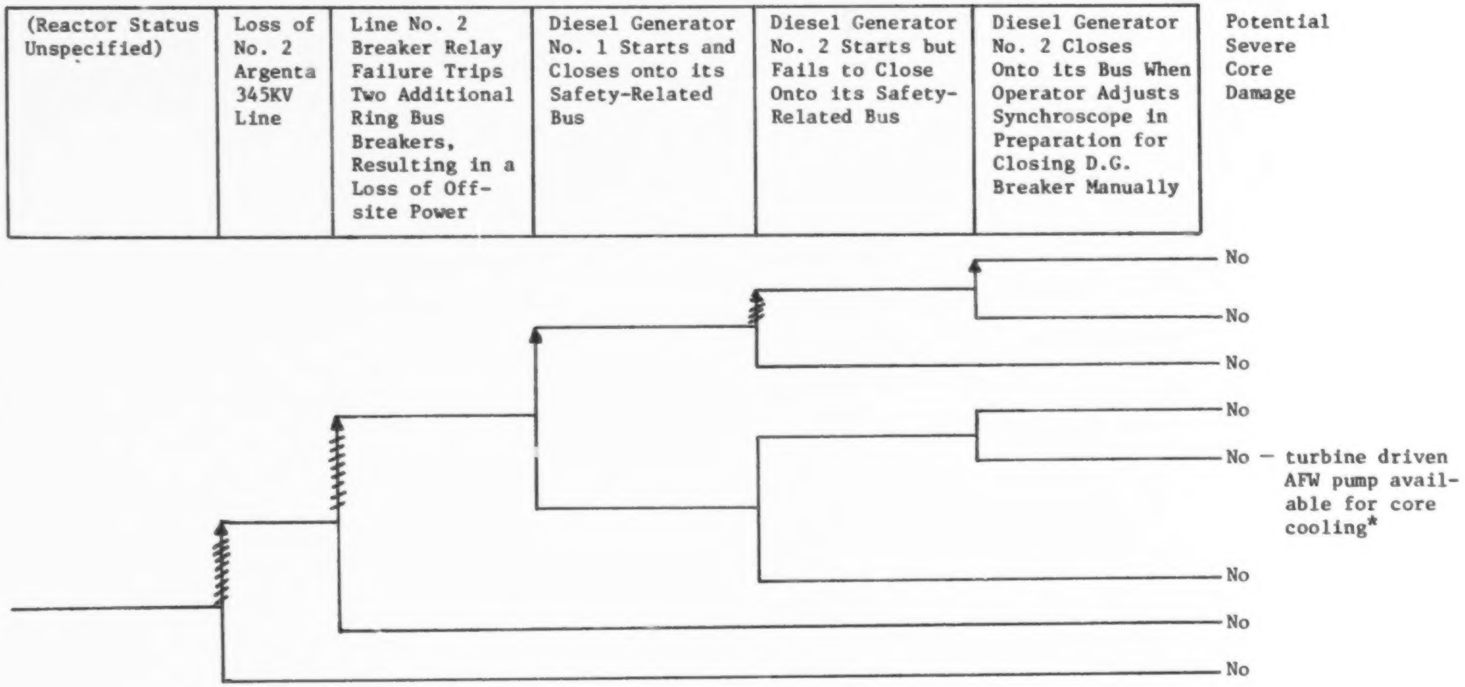
Offsite power provides electrical power to safety-related components when the unit generator is inoperable. The diesel generators provide a standby source of electric power for safety-related components when both the unit generator and the offsite power sources are not available.

Unavailability of system per WASH 1400:* **Offsite Power: 2×10^{-5}/hr**

Unavailability of component per WASH 1400:* **Diesel-generator: 3×10^{-2}/D**

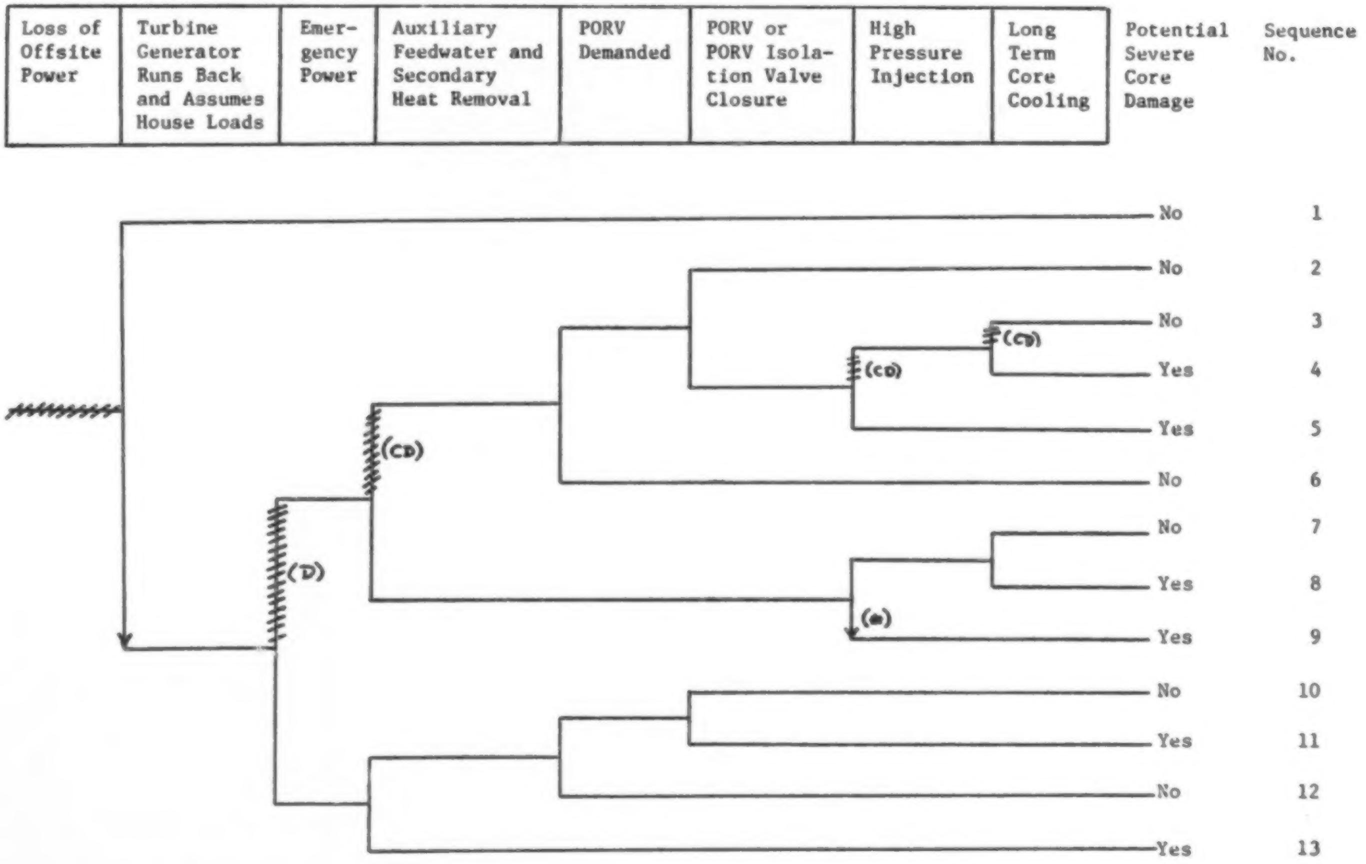
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


(Reactor Status Unspecified)
Loss of No. 2 Argenta 345KV Line
Line No. 2 Breaker Relay Failure Trips Two Additional Ring Bus Breakers, Resulting in a Loss of Off-site Power
Diesel Generator No. 1 Starts and Closes onto its Safety-Related Bus
Diesel Generator No. 2 Starts but Fails to Close Onto its Safety-Related Bus
Diesel Generator No. 2 Closes Onto its Bus When Operator Adjusts Synchroscope in Preparation for Closing D.G. Breaker Manually
Potential Severe Core Damage
No
No
No
No
No — turbine driven AFW pump available for core cooling*
No
No
No

NSIC 61565 — Actual Occurrence for Loss of Offsite Power and Failure of a Diesel Generator to Load at Palisades

*provided the turbine-driven AFW pump is not dependent on A.C. power


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
(D)
(CD)
(CD)
(CD)
(*)
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
Yes 9
No 10
Yes 11
No 12
Yes 13

NSIC 61565 — Sequence of Interest for Loss of Offsite Power and Failure of a Diesel Generator to Load at Palisades

[*] Use of HPI following AFW failures not included in mitigation procedures.

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 61565

DATE OF LER: September 9, 1971

DATE OF EVENT: September 2, 1971

SYSTEM INVOLVED: offsite power; emergency on-site power

COMPONENT INVOLVED: switchyard breaker relay, diesel generator output relay

CAUSE: failure of switchyard breaker relay, failure to close for diesel generator breaker

SEQUENCE OF INTEREST: loss of offsite power

ACTUAL OCCURRENCE: Loss of offsite power and failure of a diesel generator to load

REACTOR NAME: Palisades

DOCKET NUMBER: 50-255

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 805 MWe

REACTOR AGE: .3 yr

VENDOR: Combustion Engineering

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Consumers Power Company

LOCATION: 5 miles south of South Haven, Mich.

DURATION: N/A

PLANT OPERATING CONDITION: not known

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; [✓ (d)] failed to load.

DISCOVERY METHOD: during operation

COMMENT: –

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 63129

Date: March 24, 1971

Title: Scram Caused by Electrical Load Rejection at Lacrosse

The failure sequence was:

1. A fire in the switchyard caused the 69 kv tie line breaker to trip which resulted in a 42.5 MWe load rejection and loss of offsite power.
2. The reactor tripped on high neutron flux and the turbine was manually tripped.
3. The diesel generator started and loaded.
4. The following equipment was lost during the duration of the incident and is reported because of safety significance:
    a. The condenser circulating water pumps tripped on undervoltage. This resulted from the loop condition. (When the main condenser circulating water pumps tripped the blowdown capability of the main condenser was lost.)

(see attached sheet)

Corrective action:

1. Offsite power was restored in 61 minutes.
2. It was planned to install a cross connect valve between the high pressure service water and the low pressure service water. This would keep the heat sink for the component cooling water condensers and the decay heat removal system available.

Design purpose of failed system or component:

1. The main condenser serves as a heat sink for reactor blowdown given the need.
2. The turbine bypass valve allows steam to be dumped directly to the main condenser.
3. Low Pressure Service water provides a heat sink for the component cooling water condenser and the decay heat removal system.

Unavailability of system per WASH 1400:* Loop: 2×10^{-5}/hr

Unavailability of component per WASH 1400:* -

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

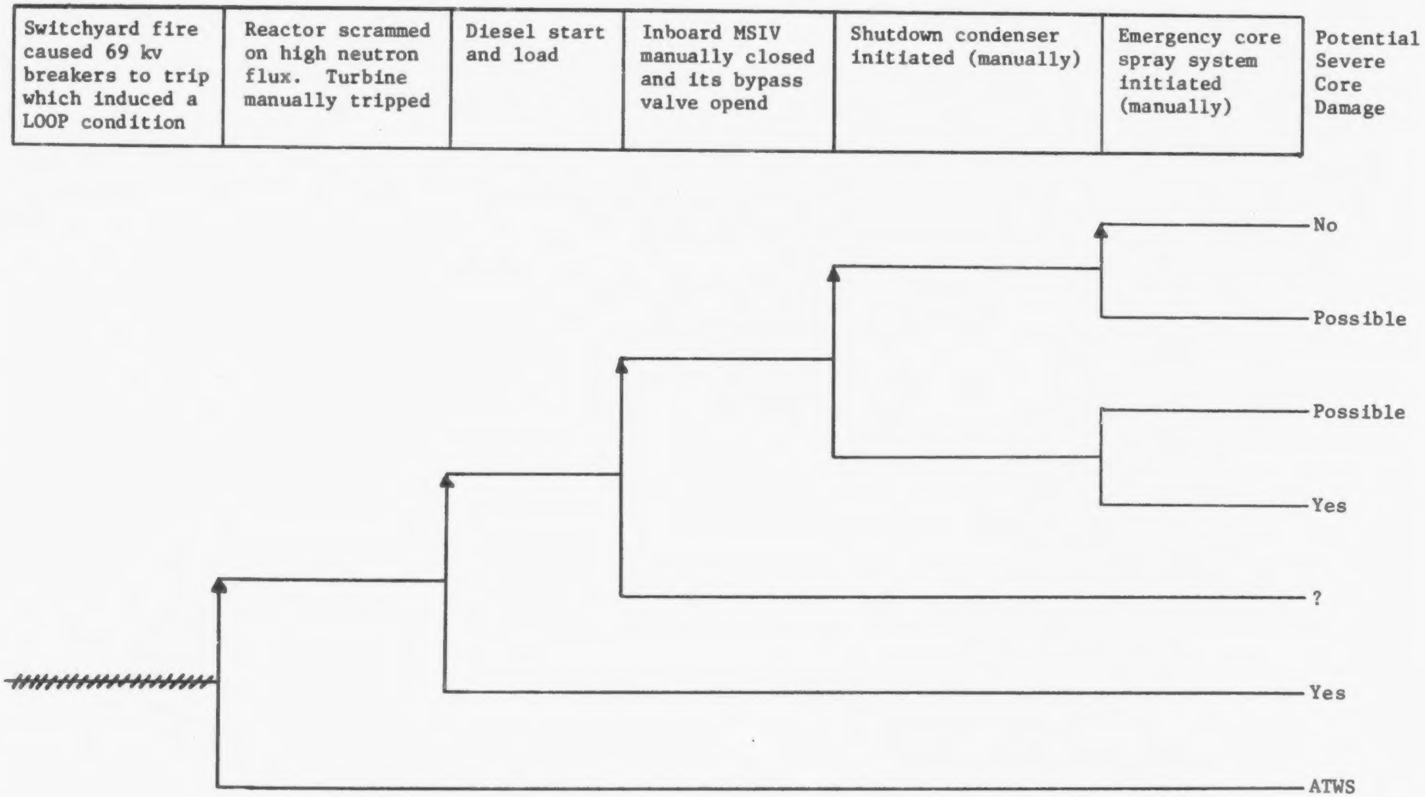
BEST DOCUMENT AVAILABLE

[redacted]

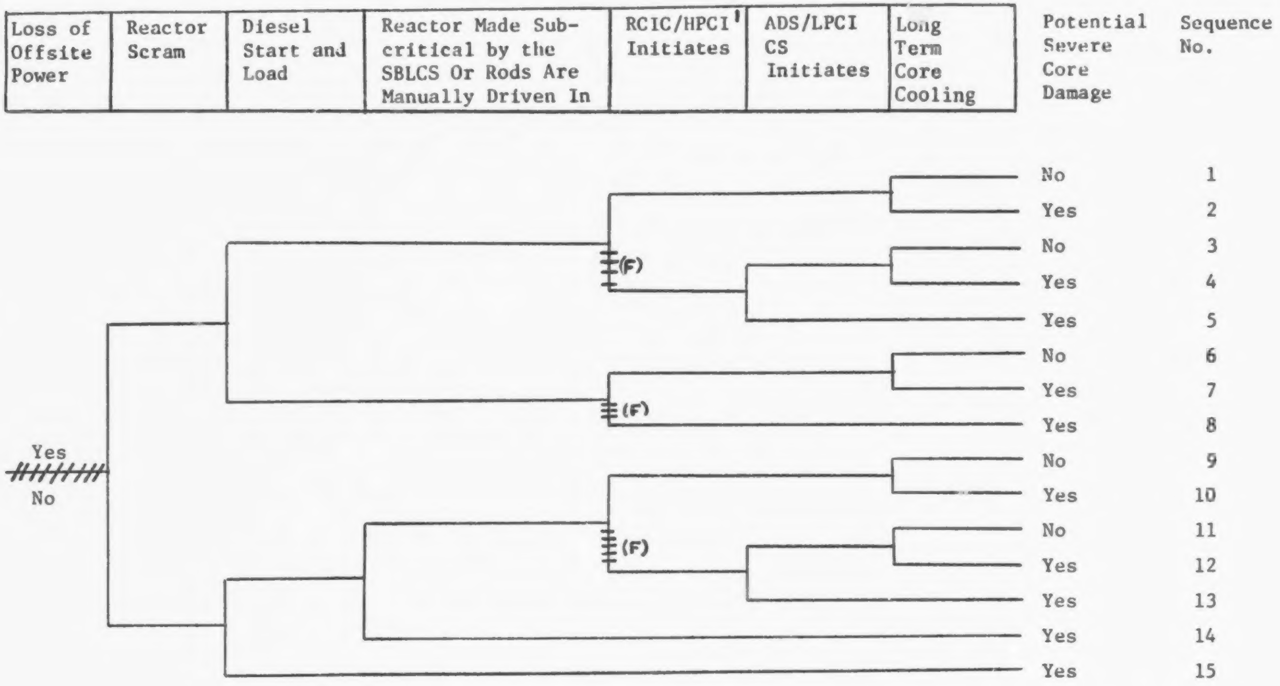
The failure sequence was: (continued)

   b. The turbine bypass valve failed because of insufficient oil pressure. The pumps providing oil pressure to the valve was supplied from non-essential a.c. power.

5. The inboard MSIVs were manually shut and their bypass valves opened for pressure control.
6. The shutdown condenser was manually initiated (same function as an isolation condition).
7. The core spray system was manually initiated.


Switchyard fire caused 69 kv breakers to trip which induced a LOOP condition
Reactor scrammed on high neutron flux. Turbine manually tripped
Diesel start and load
Inboard MSIV manually closed and its bypass valve opend
Shutdown condenser initiated (manually)
Emergency core spray system initiated (manually)
Potential Severe Core Damage
No
Possible
Possible
Yes
?
Yes
ATWS

NSIC 63129 — Actual Occurrence of A Scram Caused by Electrical Load Rejection at La Crosse


Loss of Offsite Power
Reactor Scram
Diesel Start and Load
Reactor Made Subcritical by the SBLCS Or Rods Are Manually Driven In
RCIC/HPCI[1] Initiates
ADS/LPCI CS Initiates
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
(F)
(F)
(F)
No 1
Yes 2
No 3
Yes 4
Yes 5
No 6
Yes 7
Yes 8
No 9
Yes 10
No 11
Yes 12
Yes 13
Yes 14
Yes 15

NSIC 63129 — Sequence of Interest for A Scram Caused by Electrical Load Rejection at La Crosse

[1] La Crosse utilizes shutdown condenser and condensate pumps instead of RCIC and HPCI.

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 63129

DATE OF LER: March 31, 1971

DATE OF EVENT: March 24, 1971

SYSTEM INVOLVED: Electric power

COMPONENT INVOLVED: circuit breakers

CAUSE: Fire in the switchyard.

SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE: Scram caused by electric load rejection at La Crosse

REACTOR NAME: La Crosse

DOCKET NUMBER: 50-409

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 50.0 MWe

REACTOR AGE: 4.9 yr

VENDOR: Allis Chalmers

ARCHITECT-ENGINEERS: Sargent & Lundy

OPERATORS: Dairyland Power Corporation

LOCATION: 19 miles S of LaCrosse

DURATION: N/A

PLANT OPERATING CONDITION: 90% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________________

DISCOVERY METHOD: Operational event

COMMENT: –

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 63144

Date: April 7, 1971

Title: Both Diesels Fail to Start at Robinson 2

The failure sequence was:

1. Due to a history of set point drift in the diesel generator low lube oil pressure switches, the switches were replaced with switches considered more reliable. To reduce the effects of vibration on the new switches, the switches were mounted on the room walls and connected to the diesels using approximately 15' of 1/4" copper tubing.
2. During subsequent testing, both diesels failed to run due to low indicated lube oil pressure at the pressure switches, a result of trapped air and high viscosity cold lube oil in the small tubing, which prevented transmission of the correct oil pressure to the pressure switches.

Corrective action:

The pressure switch lines were replaced with 3/8" tubing which included a vent connection.

Design purpose of failed system or component:

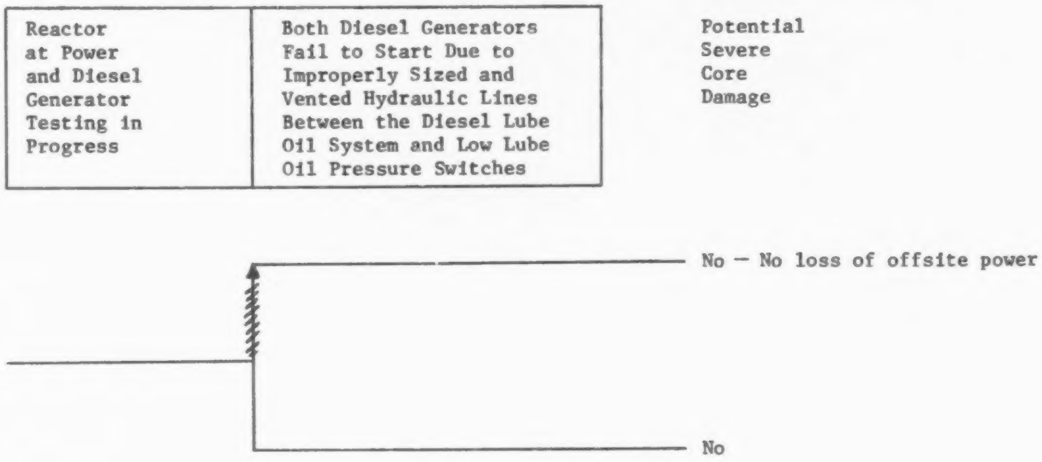
The diesel generators provide power to safety related loads in the event of unavailability of the unit generator and offsite power sources.

Unavailability of system per WASH 1400:* Emergency power: 1×10^{-2}/D

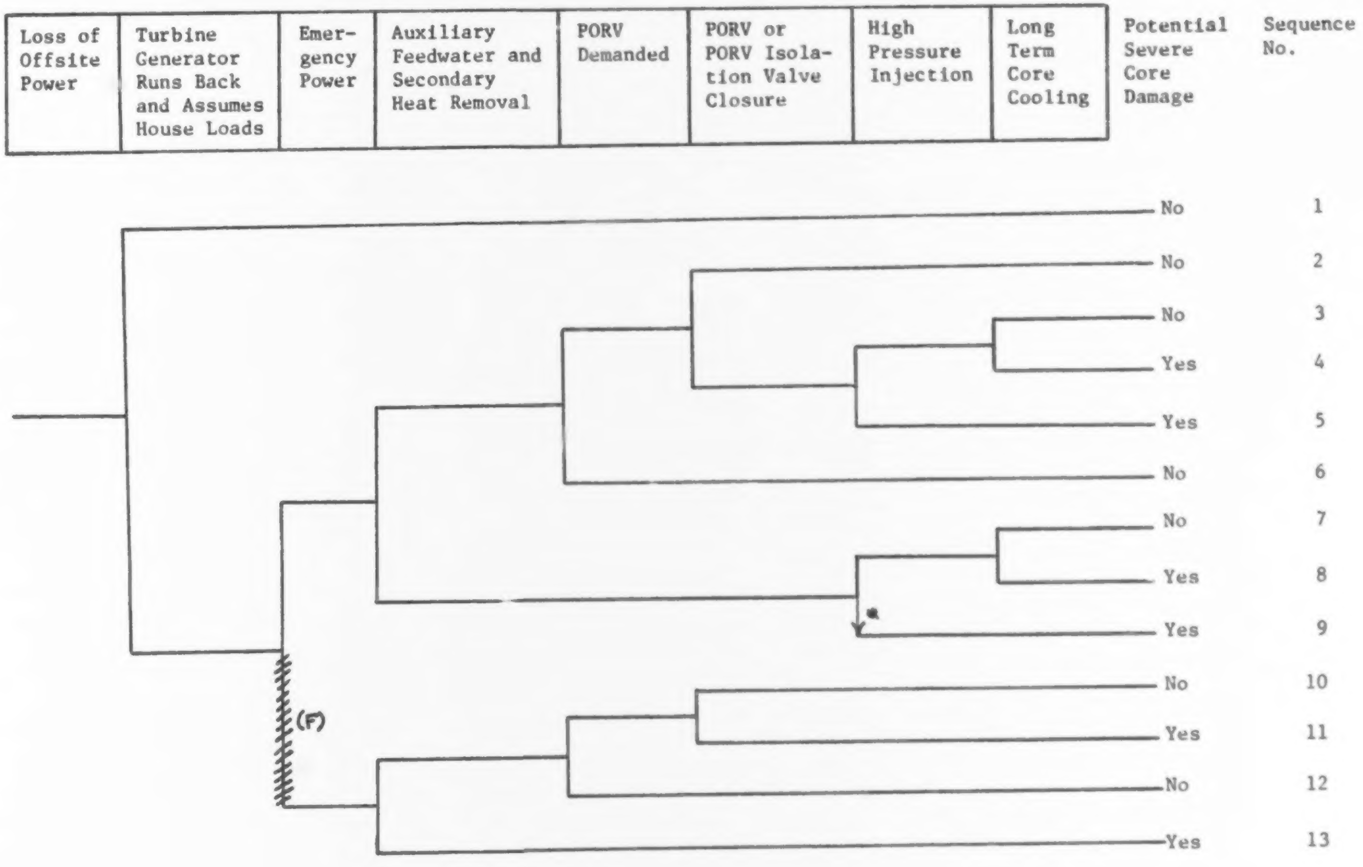
Unavailability of component per WASH 1400:* Diesel generator: 3×10^{-2}/D

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor at Power and Diesel Generator Testing in Progress
Both Diesel Generators Fail to Start Due to Improperly Sized and Vented Hydraulic Lines Between the Diesel Lube Oil System and Low Lube Oil Pressure Switches
Potential Severe Core Damage
No — No loss of offsite power
No

NSIC 63144 — Actual Occurrence for Both Diesel Generator Fail to Start at Robinson 2


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
*
Yes 9
(F)
No 10
Yes 11
No 12
Yes 13

NSIC 63144 — Sequence of Interest for Both Diesel Generators Fail to Start at Robinson 2

*not included in mitigation procedures

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 63144

DATE OF LER: April 7, 1971

DATE OF EVENT: March 8, 1971

SYSTEM INVOLVED: Emergency power

COMPONENT INVOLVED: Diesel generator low lube oil pressure switches

CAUSE: Incorrect hydraulic connection between switches and lube oil system

SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE: Diesel generator failure to start during testing

REACTOR NAME: Robinson 2

DOCKET NUMBER: 50-261

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 700 MWe

REACTOR AGE: .43 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Ebasco

OPERATORS: Carolina Power & Light Co.

LOCATION: 5 miles NW of Hartsville, SC

DURATION: 360(a) hours

PLANT OPERATING CONDITION: At power (level unspecified)

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________

DISCOVERY METHOD: Testing

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 64600

Date: June 30, 1971

Title: Loss of Offsite Power at Indian Point 1

The failure sequence was:

1. Both sources of offsite power were supplied through a single substation transformer (the normal second transformer was out-of-service for inspection).
2. The transformer tripped due to an inadvertent relay actuation, resulting in a total loss of off-site power.
3. (Since no other failures were reported it is assumed the plant tripped and emergency feedwater was utilized for reactor cooling.)

Corrective action:

1. Modified operating procedures to provide greater assurance that at least two feeders, provided from separate sources, will be available at all times.

Design purpose of failed system or component:

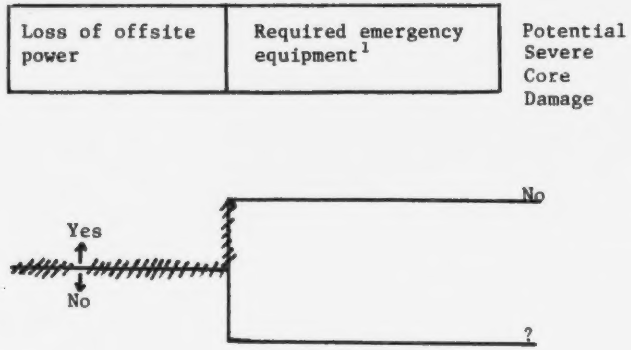
1. Provide electric power to plant components when unit generator is not in operation.

Unavailability of system per WASH 1400:* 2×10^{-5}/hr

Unavailability of component per WASH 1400:*

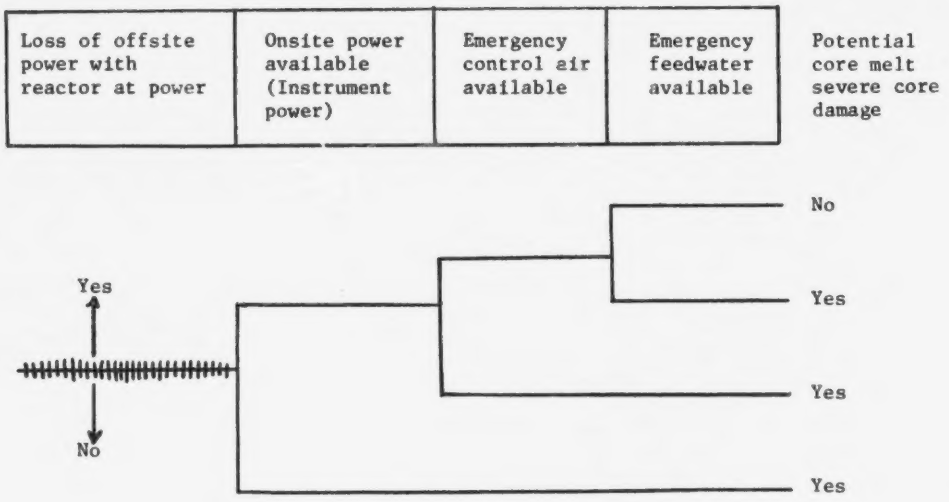
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Loss of offsite power
Required emergency equipment[1]
Potential Severe Core Damage
Yes
No
No
?

NSIC 64600 — Actual Occurrence for Loss of Offsite Power at Indian Point 1 (Operations Report No. 17)

[1] Operations report does not specify systems utilized after the loss of power occurred beyond this statement.


Loss of offsite power with reactor at power
Onsite power available (Instrument power)
Emergency control air available
Emergency feedwater available
Potential core melt severe core damage
Yes
No
No
Yes
Yes
Yes

NSIC 64600 — Sequence of Interest for Loss of Offsite Power at Indian Point 1 (Operations Report No. 17)

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 64600

DATE OF LER: June 30, 1971 (Operations Report No. 17)

DATE OF EVENT: December 26, 1970

SYSTEM INVOLVED: Off site ac power

COMPONENT INVOLVED: Transmission lines

CAUSE: Shutdown of Buchanan Substation

SEQUENCE OF INTEREST: Loss of Offsite Power

ACTUAL OCCURRENCE: Shutdown of Buchanan Station.

REACTOR NAME: Indian Point 1

DOCKET NUMBER: 50-3

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 265 MWe

REACTOR AGE: 9.0 yr

VENDOR: B&W

ARCHITECT-ENGINEERS: Consolidated Edison Co.

OPERATORS: Consolidated Edison Co.

LOCATION: 3 miles SW of Peekskill, NY

DURATION: N/A

PLANT OPERATING CONDITION: Unknown.

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c) made inoperable]; (d) ________________

DISCOVERY METHOD: Operational event.

COMMENT: –

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 65757

Date: June 30, 1971

Title: Loss of Offsite Power at Point Beach 1 (Semi-Annual Operations Report for January—June, 1971)

The failure sequence was:

1. With the reactor at hot standby during an ice storm, breakers on all three high lines opened (reason unspecified).
2. A reactor trip occurred.
3. Both diesel generators started and supplied safety-related loads.
4. In view of the continuing storm conditions, the reactor coolant system was borated to the cold shutdown level and cooled down to 300°F.

Corrective action:

The unit was returned to power two days after the event.

Design purpose of failed system or component:

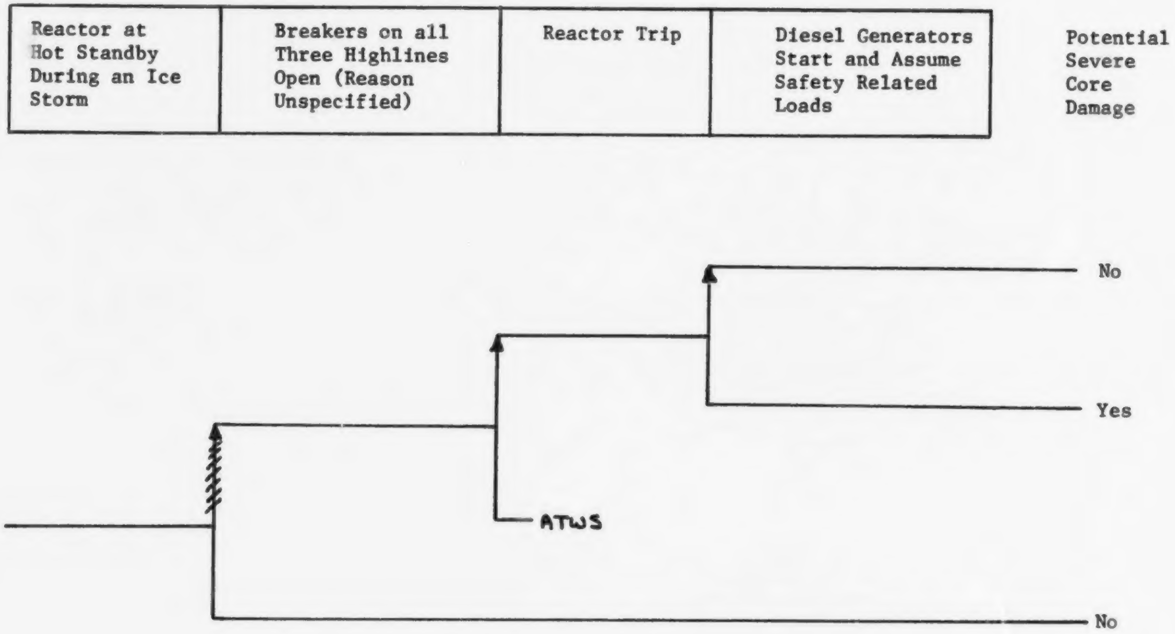
Offsite power is the preferred source of power to safety related loads when the unit generator is not producing power.

Unavailability of system per WASH 1400:* Loss of offsite power: 2×10^{-5}/hr

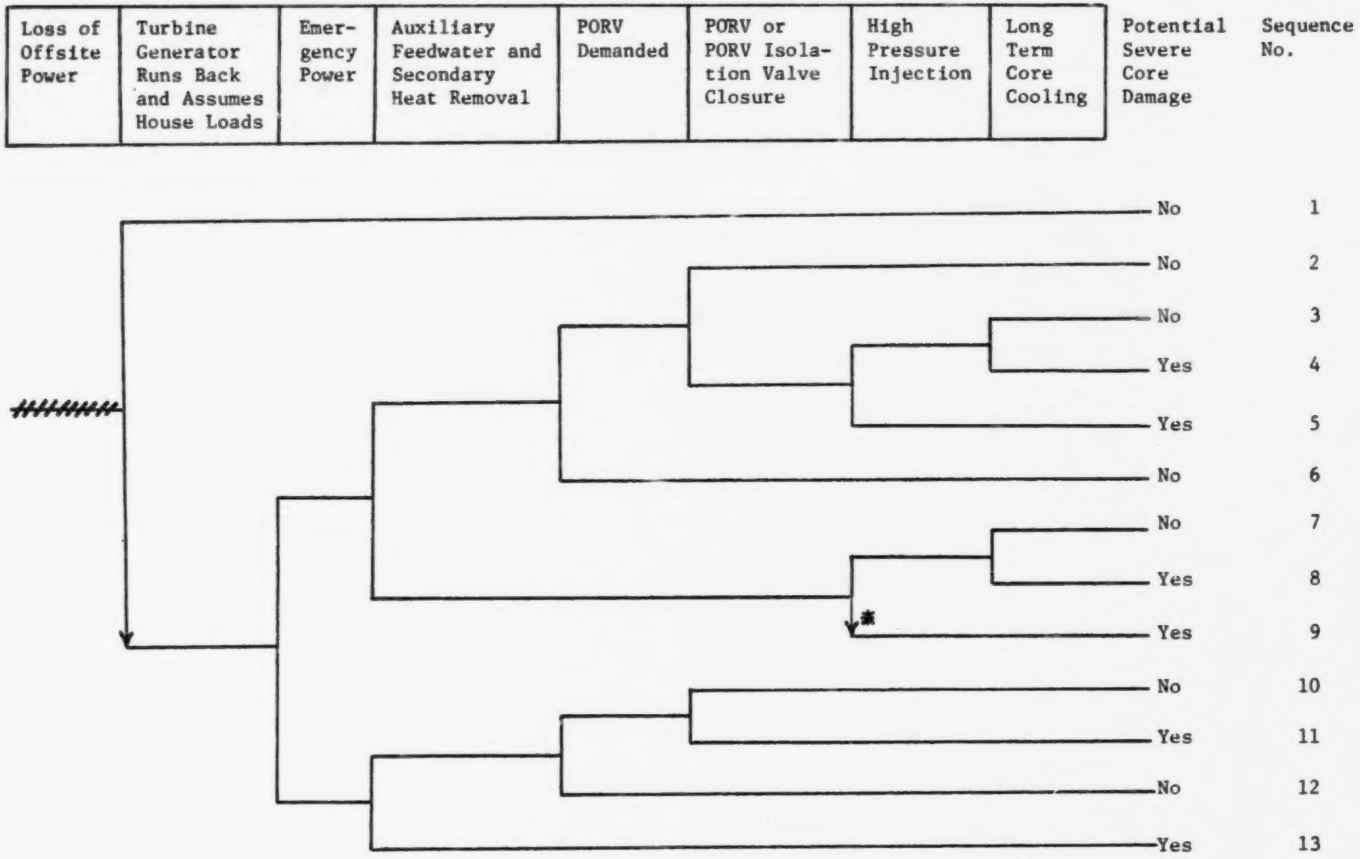
Unavailability of component per WASH 1400:* N/A

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor at Hot Standby During an Ice Storm
Breakers on all Three Highlines Open (Reason Unspecified)
Reactor Trip
Diesel Generators Start and Assume Safety Related Loads
Potential Severe Core Damage
No
Yes
ATWS
No

NSIC 65757 — Actual Occurrence for Loss of Offsite Power at Point Beach 1


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
*
Yes 9
No 10
Yes 11
No 12
Yes 13

NSIC 65757 — Sequence of Interest for Loss of Offsite Power at Point Beach 1

*Not included in operating procedures

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 65757

DATE OF LER: June 30, 1971

DATE OF EVENT: February 5, 1971

SYSTEM INVOLVED: Offsite power

COMPONENT INVOLVED: Highline breakers

CAUSE: Opened (reason unspecified)

SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE: Loss of offsite power

REACTOR NAME: Point Beach 1

DOCKET NUMBER: 50-266

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 497 MWe

REACTOR AGE: .26 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Wisconsin Electric Power Co.

LOCATION: 15 miles north of Manitowoc, Wisc.

DURATION: N/A

PLANT OPERATING CONDITION: Hot standby

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ______________

DISCOVERY METHOD: Operational event

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 65757

Date: June 30, 1971

Title: Failure of Containment Sump Isolation Valves at Point Beach 1
(Semi-Annual Operations Report for January-June, 1971)

The failure sequence was:

1. During a reactor heatup, the control room operator attempted to open both containment sump isolation valves.
2. Both valves failed to open due to air-binding in their hydraulic operating mechanisms. In addition, both hydraulic units were found to be installed 90% from their upright positions.

Corrective action:

The valve hydraulic mechanisms were re-oriented and the suction piping re-routed to take suction from the bottom of the oil sumps.

Design purpose of failed system or component:

The valves isolate the containment sump from the low pressure injection/recirculation system during normal operation. After the RWST is emptied following a LOCA, these valves are opened to provide suction flow to the low pressure recirculation system.

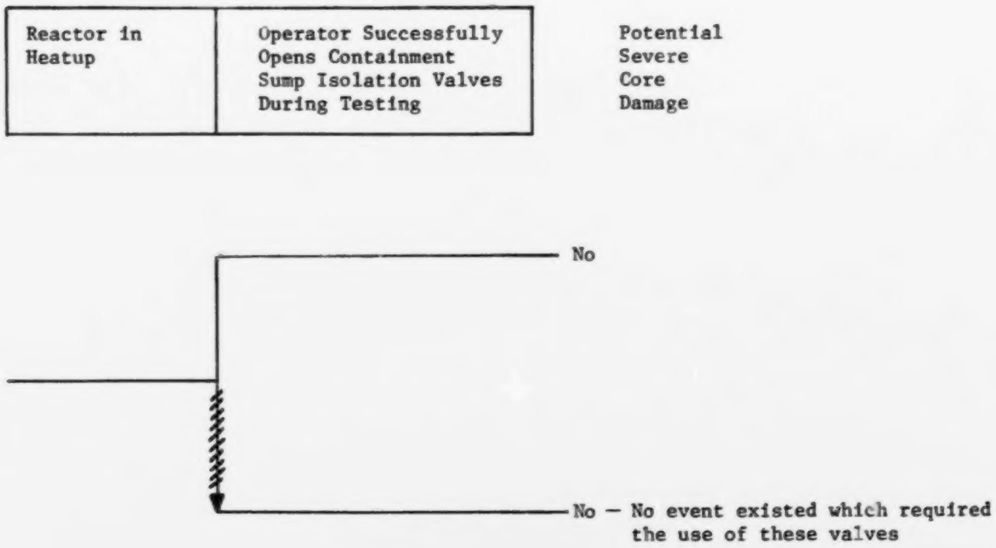
Unavailability of system per WASH 1400:* Low pressure recirculation: $1.3 \times 10^{-2}/D$

Unavailability of component per WASH 1400:* Valves, motor operated, failure to operate: $1 \times 10^{-3}/D$

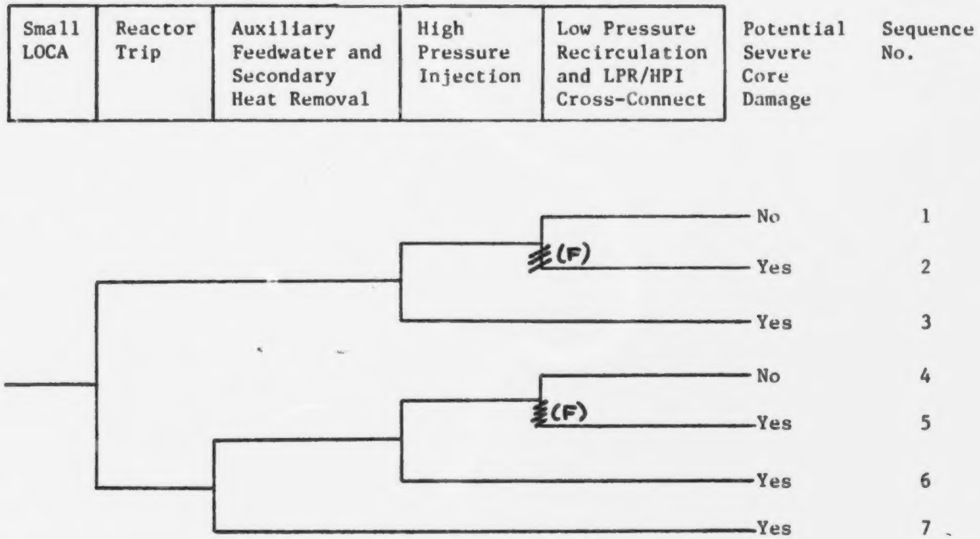
Valves, air-fluid operated, failure to operate: $3 \times 10^{-4}/D$

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor in Heatup
Operator Successfully Opens Containment Sump Isolation Valves During Testing
Potential Severe Core Damage
No
No — No event existed which required the use of these valves

NSIC 65757 — Actual Occurrence for Failure of Containment Sump Isolation Valves at Point Beach 1


Small LOCA
Reactor Trip
Auxiliary Feedwater and Secondary Heat Removal
High Pressure Injection
Low Pressure Recirculation and LPR/HPI Cross-Connect
Potential Severe Core Damage
Sequence No.
(F)
(F)
No
Yes
Yes
No
Yes
Yes
Yes
1
2
3
4
5
6
7

NSIC 65757 — Sequence of Interest for Failure of Containment Sump Isolation Valves at Point Beach 1

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 65757

DATE OF LER: June 30, 1971

DATE OF EVENT: April 16, 1971

SYSTEM INVOLVED: Low pressure recirculation

COMPONENT INVOLVED: Containment sump isolation valves

CAUSE: Valves failed closed due to air-bound hydraulic operators

SEQUENCE OF INTEREST: LOCA

ACTUAL OCCURRENCE: Sump isolation valves failed closed during testing

REACTOR NAME: Point Beach 1

DOCKET NUMBER: 50-266

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 497 MWe

REACTOR AGE: .45 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Wisconsin Electric Power Co.

LOCATION: **15 miles north of Manitowoc, Wisc.**

DURATION: 360(a) hours

PLANT OPERATING CONDITION: Reactor heatup

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; (c) made inoperable; [✓ (d)] failed to open

DISCOVERY METHOD: Testing

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 65969

Date: September 16, 1971

Title: Unclear Wiring Diagrams Result in Depressurization at Palisades

The failure sequence was:

1. With the plant in hot shutdown, a technician de-energized breakers to the reactor protective system. This resulted in loss of power to the electromatic relief valve pilot valve solenoids opening the valves.
2. One of the electromatic relief valves was isolated because its isolation valve was closed; however, the open, unisolated relief valve permitted RCS blowdown to the quench tank.
3. Safety injection was initiated on both safety injection channels, however channel A was blocked by the operator.
4. The operator closed the electromatic relief valve isolation valve and started the third charging pump.

(Continued on attached sheet)

Corrective action:

1. The reactor protective system drawings were to be corrected to indicate the "as-built" plant condition using standard notation.
2. The electromatic relief valve control scheme was to be reviewed to determine if any changes were desirable to lessen the probability of a second incident.

Design purpose of failed system or component:

1. The electromatic relief valves provide RCS overpressure protection at a pressure below the RCS safety valve set points.

Unavailability of system per WASH 1400:* —

Unavailability of component per WASH 1400:* relief valve, failure to reseat: 10^{-2}/D

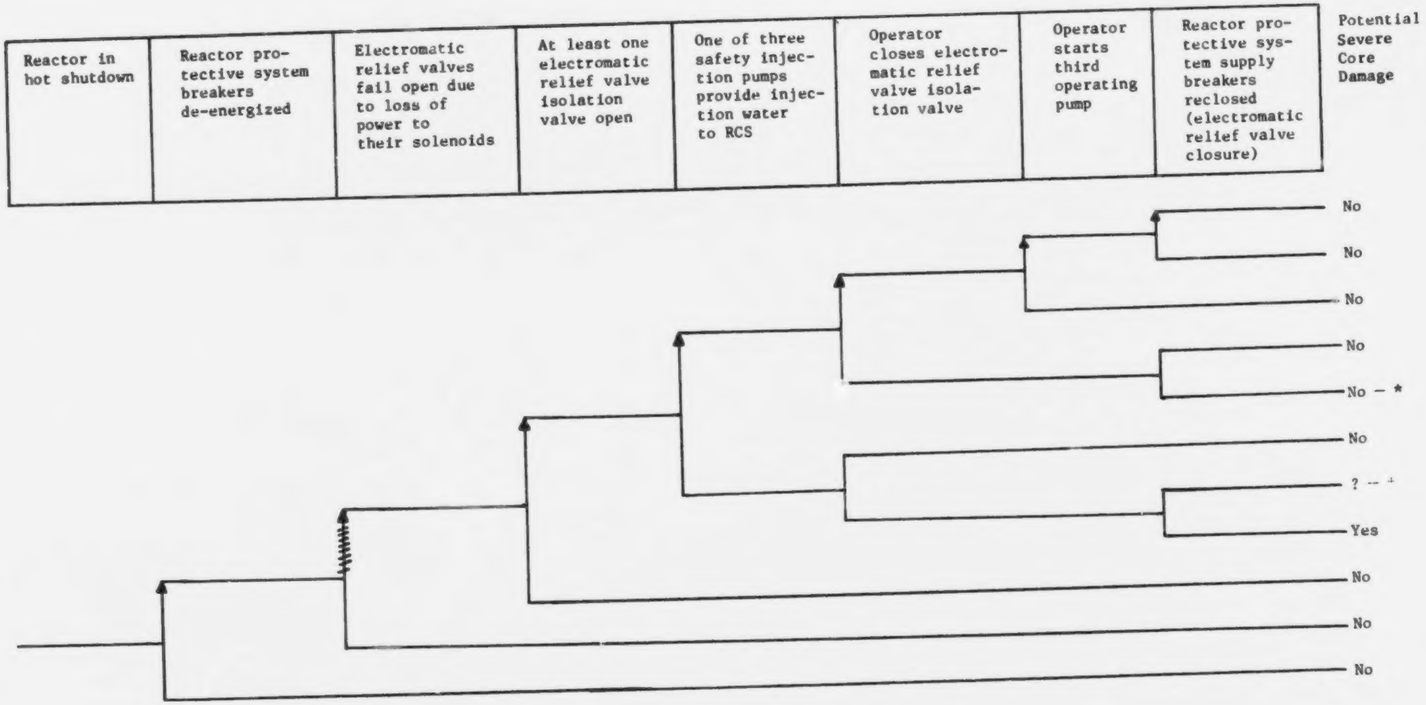
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

[redacted]

The failure sequence was: (continued)

5. The RCS depressurized to 1280 psia in approximately two minutes.
6. The reactor protective system breakers were reclosed, restoring power to the electromatic relief valve solenoids and thus closing the relief valves.

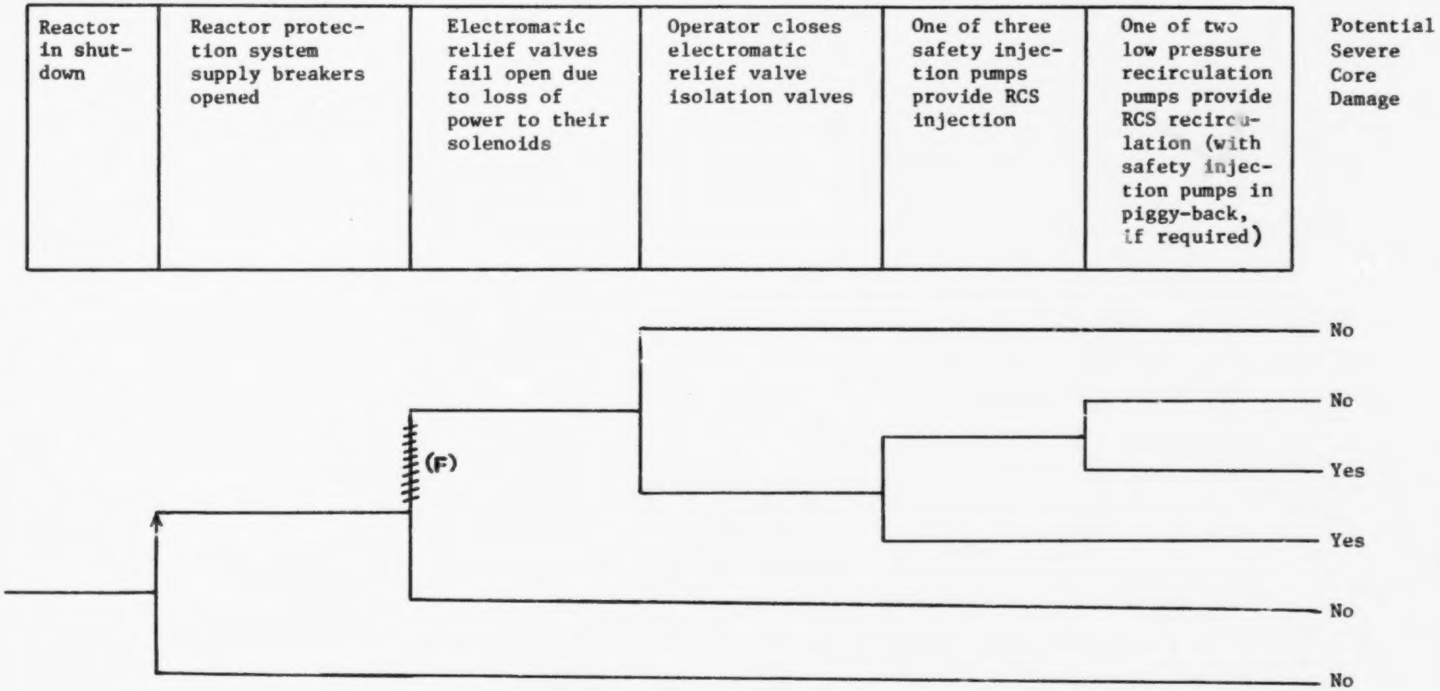

Reactor in hot shutdown
Reactor protective system breakers de-energized
Electromatic relief valves fail open due to loss of power to their solenoids
At least one electromatic relief valve isolation valve open
One of three safety injection pumps provide injection water to RCS
Operator closes electromatic relief valve isolation valve
Operator starts third operating pump
Reactor protective system supply breakers reclosed (electromatic relief valve closure)
Potential Severe Core Damage
No
No
No
No
No — *
No
? — +
Yes
No
No
No

NSIC 65969 — Actual Occurrence for Unclear Wiring Diagrams Result in Depressurization at Palisades

---

*safety injection available for small break mitigation

+8 minute RCS blowdown, system would have been saturated


Reactor in shutdown
Reactor protection system supply breakers opened
Electromatic relief valves fail open due to loss of power to their solenoids
Operator closes electromatic relief valve isolation valves
One of three safety injection pumps provide RCS injection
One of two low pressure recirculation pumps provide RCS recirculation (with safety injection pumps in piggy-back, if required)
Potential Severe Core Damage
(F)
No
No
Yes
Yes
No
No

NSIC 65969 – Sequence of Interest for Unclear Wiring Diagrams Result in Depressurization at Palisades

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 65969

DATE OF LER: September 16, 1971

DATE OF EVENT: September 8, 1971

SYSTEM INVOLVED: Reactor Coolant System, Reactor Protective System

COMPONENT INVOLVED: Electromatic Relief Valves

CAUSE: Opening of reactor protective system supply breakers resulted in opening of electromatic relief valves

SEQUENCE OF INTEREST: small LOCA (open electromatic relief valve)

ACTUAL OCCURRENCE: open electromatic relief valve

REACTOR NAME: Palisades

DOCKET NUMBER: 50-255

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 805 MWe

REACTOR AGE: .3 yr

VENDOR: Combustion Engineering

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Consumers Power Co.

LOCATION: 5 miles south of South Haven, Mich.

DURATION: N/A

PLANT OPERATING CONDITION: hot shutdown

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; (c) made inoperable; [✓ (d)] failed open

DISCOVERY METHOD: During operation

COMMENT: –

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 66996

Date: October 10, 1971

Title: Pressure Transient and Blowdown at Millstone Point 1

The failure sequence was:

1. A malfunction in the turbine pressure control system (not specified) caused a pressure transient which resulted in a reactor trip on high neutron flux.
2. The turbine was manually tripped. This caused the turbine bypass valve to open as expected.
3. The #1 bypass valve failed to close so the operator manually closed the MSIVs.
4. The blowdown continued through an open relief valve until the reactor pressure reached 263 psig when it reseated. The valve's main disk was wire drawn and badly eroded.

(See attached page)

Corrective action:

1. The relief valve was ground,rebuilt, and tested prior to being reinstalled.
2. The turbine control problem was thought to be with the linkage however this was being investigated.

Design purpose of failed system or component:

Relief valves provide for reactor pressure control

Unavailability of system per WASH 1400:* Relief valve failure to reset: $1 \cdot 10^{-2}/D$

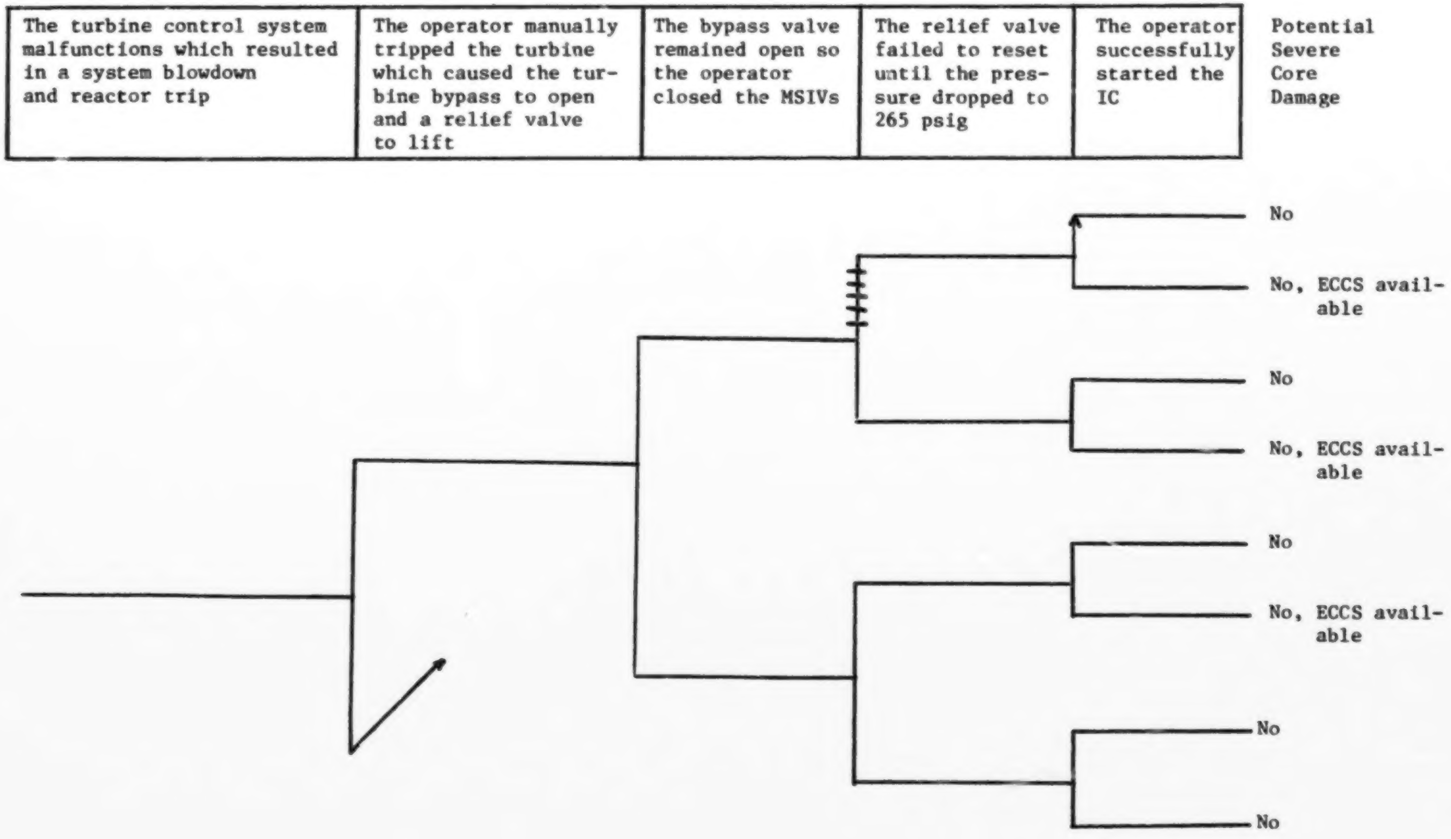
Unavailability of component per WASH 1400:*

---

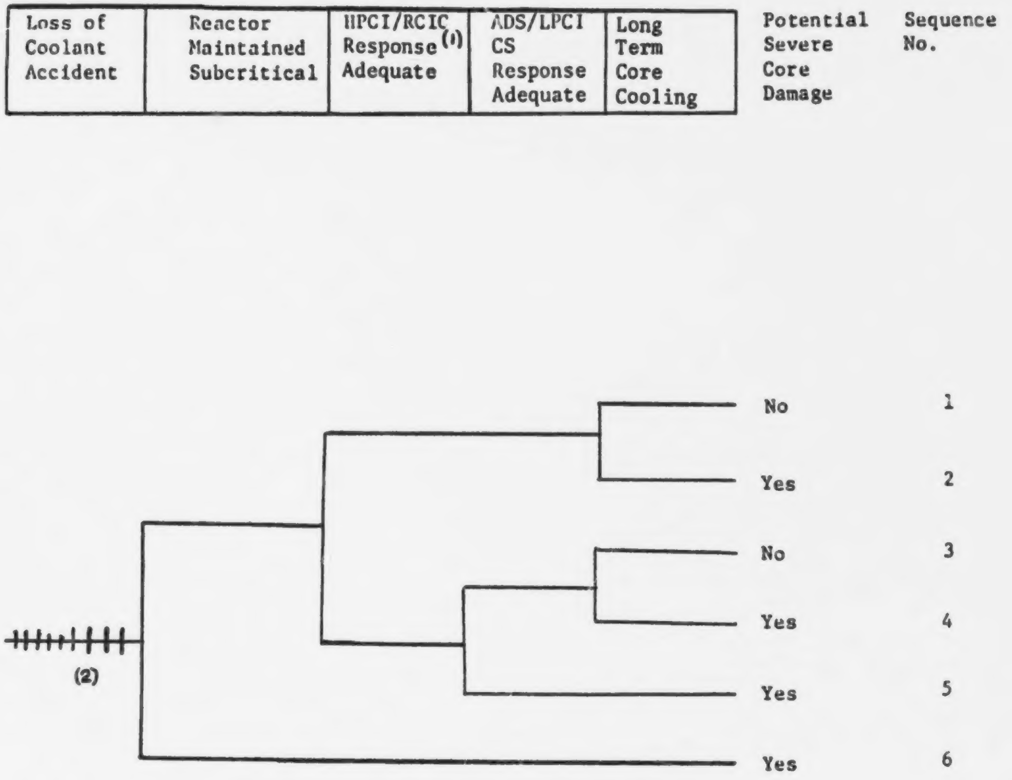
*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

[redacted]

5. The operator initiated the isolation condenser and proceeded with a controlled cooldown. A total of 75,000 gallons of water was lifted from the torus, however it was not stated how the water was injected into the core.


The turbine control system malfunctions which resulted in a system blowdown and reactor trip
The operator manually tripped the turbine which caused the turbine bypass to open and a relief valve to lift
The bypass valve remained open so the operator closed the MSIVs
The relief valve failed to reset until the pressure dropped to 265 psig
The operator successfully started the IC
Potential Severe Core Damage
No
No, ECCS available
No
No, ECCS available
No
No, ECCS available
No
No

NSIC 66996 — Actual Occurrence of a Pressure Transient and Blowdown at Millstone Point 1


Loss of Coolant Accident
Reactor Maintained Subcritical
HPCI/RCIC Response(1) Adequate
ADS/LPCI CS Response Adequate
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
(2)
No
1
Yes
2
No
3
Yes
4
Yes
5
Yes
6

NSIC 66996 — Sequence of Interest for Pressure Transient and Blowdown at Millstone Point 1

[1] Millstone Point 1 utilizes IC, and FWCI instead of RCIC and HPCI.

[2] Stuck open relief valve.

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 66996

DATE OF LER: October 22, 1971

DATE OF EVENT: October 10, 1971

SYSTEM INVOLVED: Turbine control system, pressure relief

COMPONENT INVOLVED: Relief valve

CAUSE: erosion and wire drawn seat

SEQUENCE OF INTEREST: Loss of Coolant Accident (small)

ACTUAL OCCURRENCE: Pressure transient and blowdown at Millstone Point 1

REACTOR NAME: Millstone Point 1

DOCKET NUMBER: 50-245

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 660 MWe

REACTOR AGE: 1 yr

VENDOR: GE

ARCHITECT-ENGINEERS: Ebasco

OPERATORS: North East Nuclear Energy Company

LOCATION: 5 miles SW of New London, Conn.

DURATION: N/A

PLANT OPERATING CONDITION: 680MWe

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; (c) made inoperable; [✓ (d)] failed to close.

DISCOVERY METHOD: Operational event

COMMENT: -

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 71694

Date: May 26, 1972

Title: Loss of Offsite Power at Palisades

The failure sequence was:

1. With the plant in a hot standby condition, the left channel safety injection system test button was pushed to initiate a quarterly test.
2. This resulted in a loss of offsite power due to the spurious operation of a differential relay on the 1-2 startup transformer. The actuation of the relay was due to unbalanced sensing currents from a current transformer — a result of the incompatibility of the installed current transformer with the 345 KV - 2.4 KV step-down situation.

Corrective action:

The differential relays were removed from startup transformers 1-1 and 1-2. High-side overcurrent and instantaneous overcurrent relays were installed.

Design purpose of failed system or component:

The differential relays provided overcurrent protection for the startup transformer.

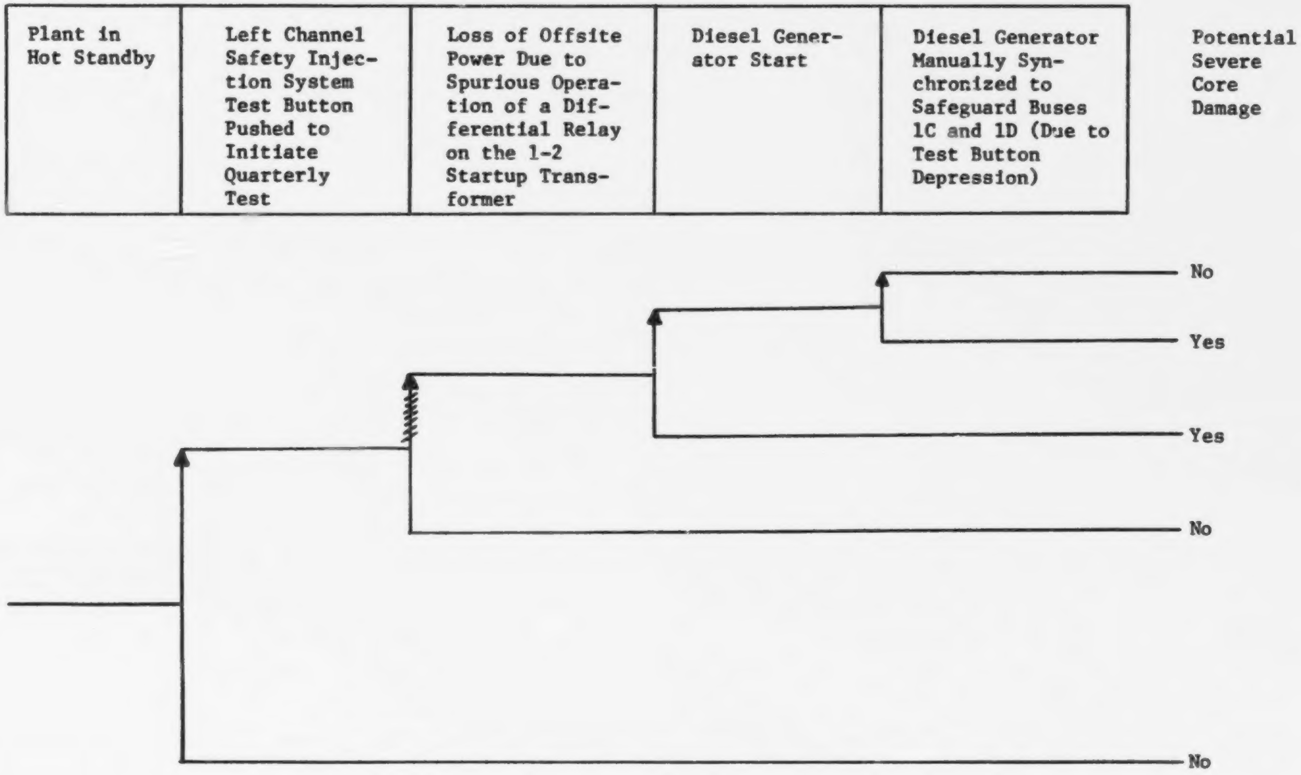
Unavailability of system per WASH 1400:* Offsite power: 2×10^{-5}/hr

Unavailability of component per WASH 1400:* —

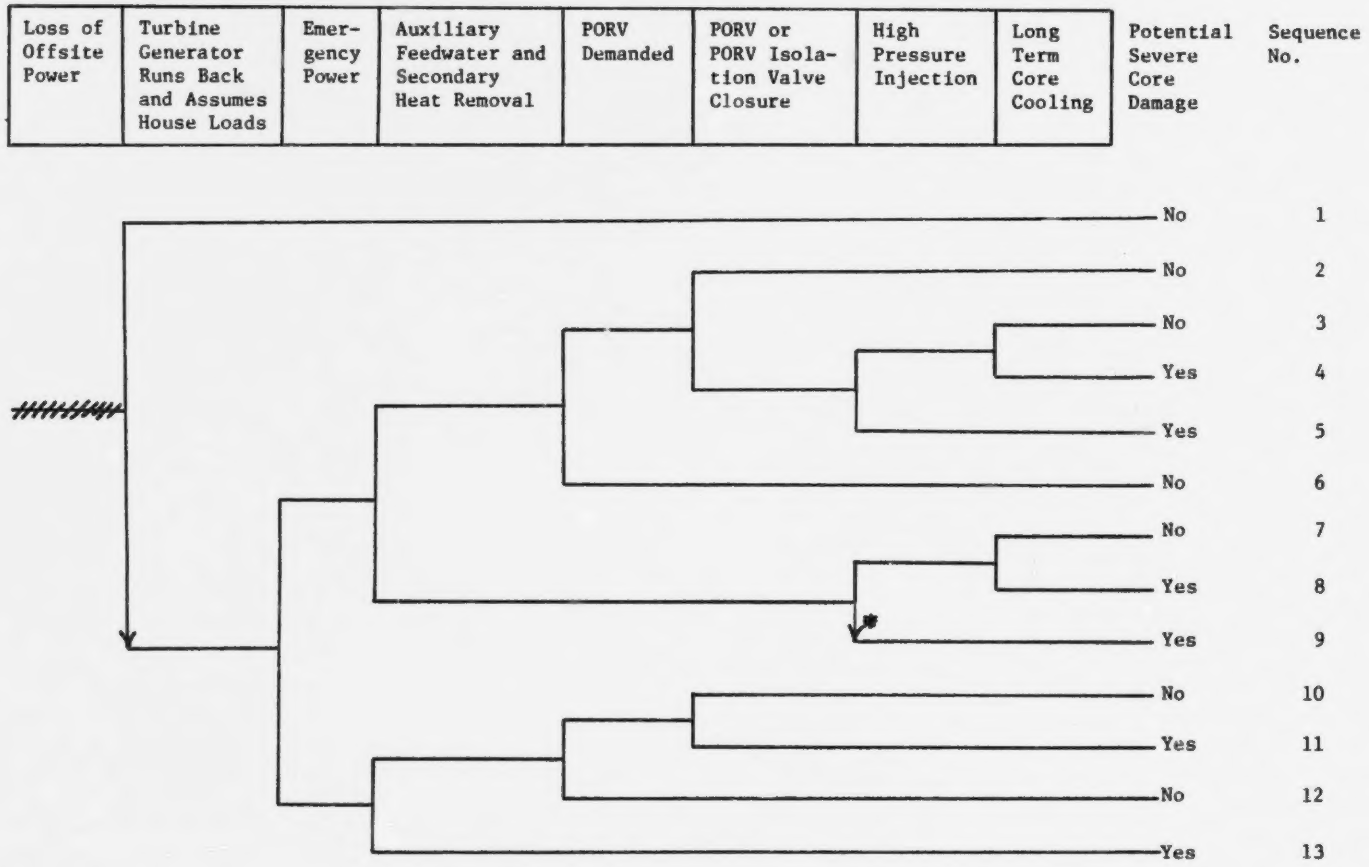
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

BEST DOCUMENT AVAILABLE


Plant in Hot Standby
Left Channel Safety Injection System Test Button Pushed to Initiate Quarterly Test
Loss of Offsite Power Due to Spurious Operation of a Differential Relay on the 1-2 Startup Transformer
Diesel Generator Start
Diesel Generator Manually Synchronized to Safeguard Buses 1C and 1D (Due to Test Button Depression)
Potential Severe Core Damage
No
Yes
Yes
No
No

NSIC 71694 — Actual Occurrence for Loss of Offsite Power at Palisades



Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
*
Yes 9
No 10
Yes 11
No 12
Yes 13

NSIC 71694 — Sequence of Interest for Loss of Offsite Power at Palisades

*Not included in operating procedures

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 71694

DATE OF LER: May 26, 1972

DATE OF EVENT: May 17, 1972

SYSTEM INVOLVED: Offsite power

COMPONENT INVOLVED: Startup transformer

CAUSE: Improperly chosen differential relay

SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE: Loss of offsite power during testing

REACTOR NAME: Palisades

DOCKET NUMBER: 255

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 805 MWe

REACTOR AGE: 1.0 yr

VENDOR: CE

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Consumers Power Corp.

LOCATION: 5 miles south of South Haven, Michigan

DURATION: N/A

PLANT OPERATING CONDITION: Hot standby

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________

DISCOVERY METHOD: During testing

COMMENT: The incompatible current transformer had been installed prior to criticality

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 73655

Date: August 18, 1972

Title: Loss of Offsite Power at Indian Point 1

The failure sequence was:

1. Reactor trip, RCP trip, MFW pump trip due to reduced voltage on plant buses due to external causes (grid disturbance).
2. Complete loss of offsite power.

Corrective action;

1. An emergency air compressor was started to provide control air.
2. Steam driven emergency feedwater pumps were used to provide steam generator cooling. Reactor cooling was provided by natural circulation.

Design purpose of failed system or component:

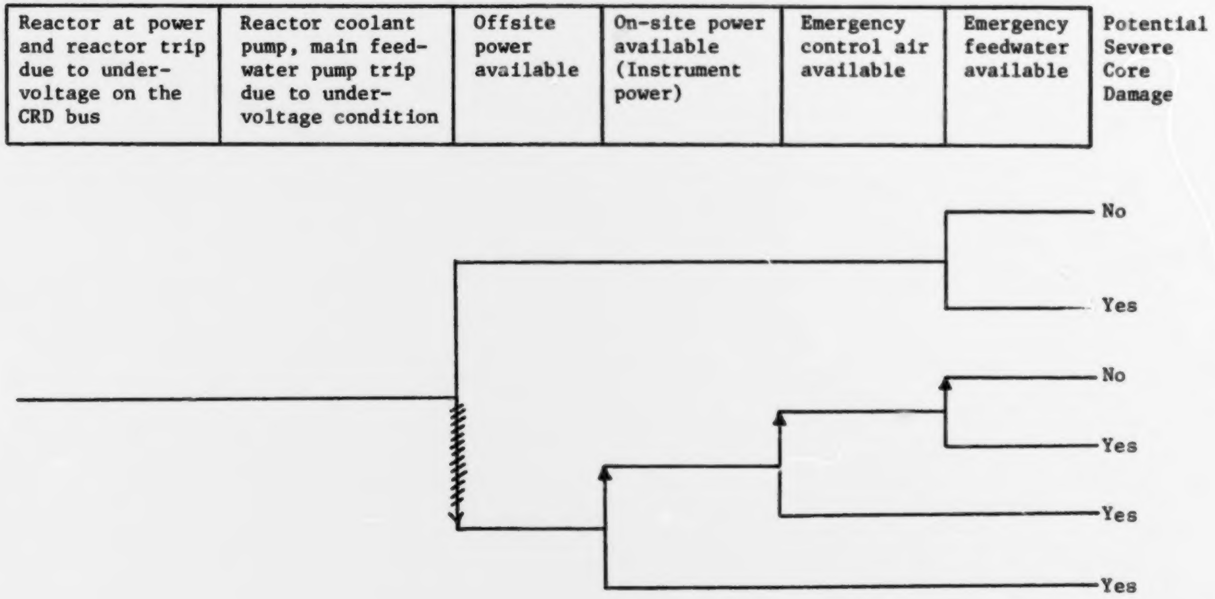
1. Offsite power provided the only source of electric power to plant components when the unit generator was not operating.

Unavailability of system per WASH 1400:* loss of offsite power: 2×10^{-5}/hr

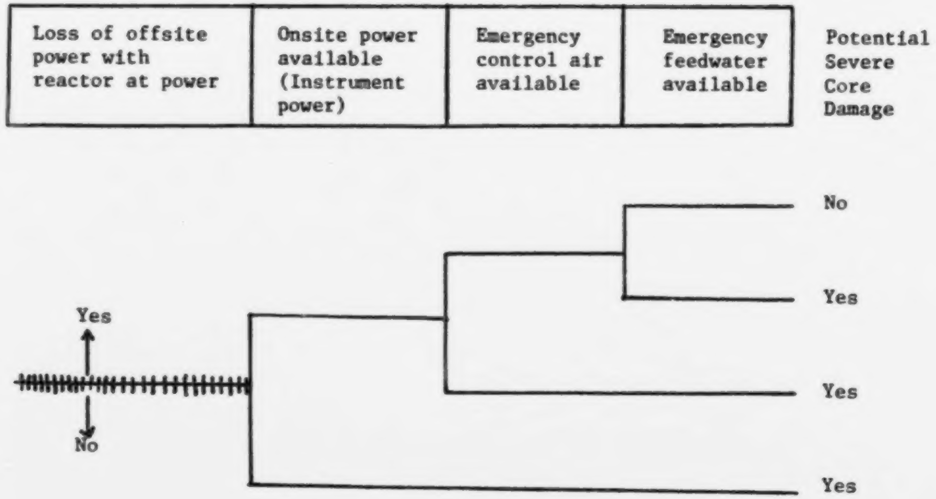
Unavailability of component per WASH 1400:*

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor at power and reactor trip due to under-voltage on the CRD bus
Reactor coolant pump, main feed-water pump trip due to under-voltage condition
Offsite power available
On-site power available (Instrument power)
Emergency control air available
Emergency feedwater available
Potential Severe Core Damage
No
Yes
No
Yes
Yes
Yes

NSIC 73655 — Actual Occurrence for Loss of Offsite Power at Indian Point 1


Loss of offsite power with reactor at power
Onsite power available (Instrument power)
Emergency control air available
Emergency feedwater available
Potential Severe Core Damage
Yes
No
No
Yes
Yes
Yes

NSIC 73655 — Sequence of Interest for Loss of Offsite Power at Indian Point 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 73655

DATE OF LER: August 18, 1972

DATE OF EVENT: July 20, 1972

SYSTEM INVOLVED: Off-site ac power

COMPONENT INVOLVED: Transmission lines

CAUSE: Undervoltage condition

SEQUENCE OF INTEREST: Loss of Offsite Power

ACTUAL OCCURRENCE: Undervoltage condition.

REACTOR NAME: Indian Point 1

DOCKET NUMBER: 50-3

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 265 MWe

REACTOR AGE: 10.4 yr

VENDOR: B&W

ARCHITECT-ENGINEERS: Consolidated Edison Co.

OPERATORS: Consolidated Edison Co.

LOCATION: 3 miles SW of Peekskill, NY

DURATION: N/A

PLANT OPERATING CONDITION: 95% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ________________

DISCOVERY METHOD: Operational event.

COMMENT: —

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 74242

Date: December 31, 1971

Title: High Coolant Level at Nine Mile Point

The failure sequence was:

1. Surveillance testing of the reactor protection system high/low water sensors was being conducted. The sensor support was accidently bumped causing a turbine trip and an anticipatory reactor trip.
2. Following the trip the reactor water level began to decrease rapidly, causing the feedwater system to respond by overfeeding, as it should, in the automatic mode.
3. After 20 seconds the operator took manual control since in his opinion the feedwater flow was high.
4. The manual transfer of control was slow and some water spilled into the main steamline.

Corrective action:
Since the automatic response of the feedwater system was correct, a review of the expected system response to transients was given to the operating staff.

Design purpose of failed system or component:
The feedwater system is part of the steam cycle and provides makeup water to the nuclear boiler.

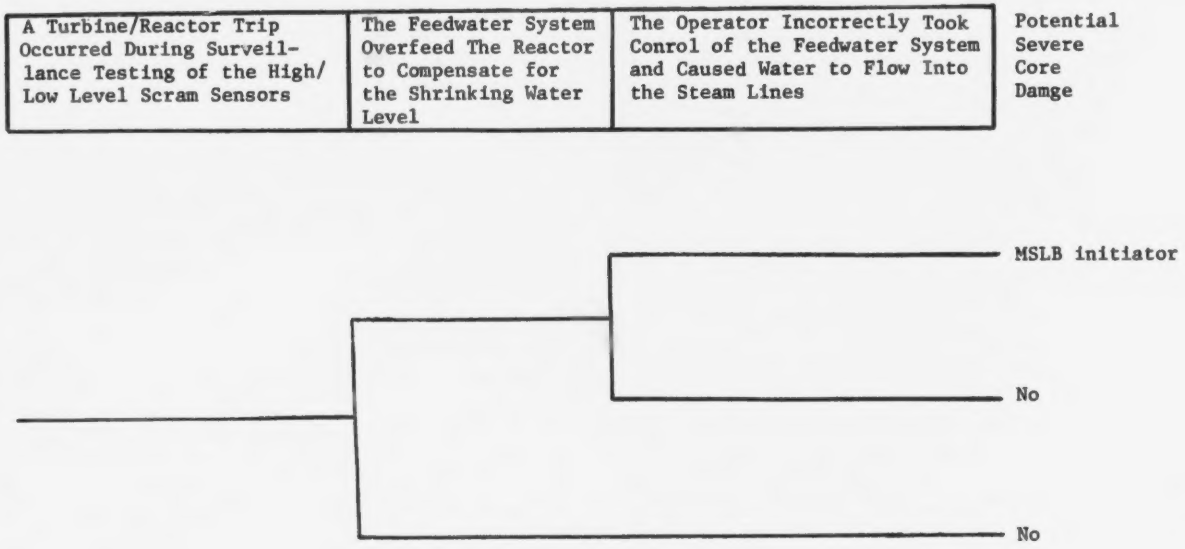
Unavailability of system per WASH 1400:* -

Unavailability of component per WASH 1400:* -

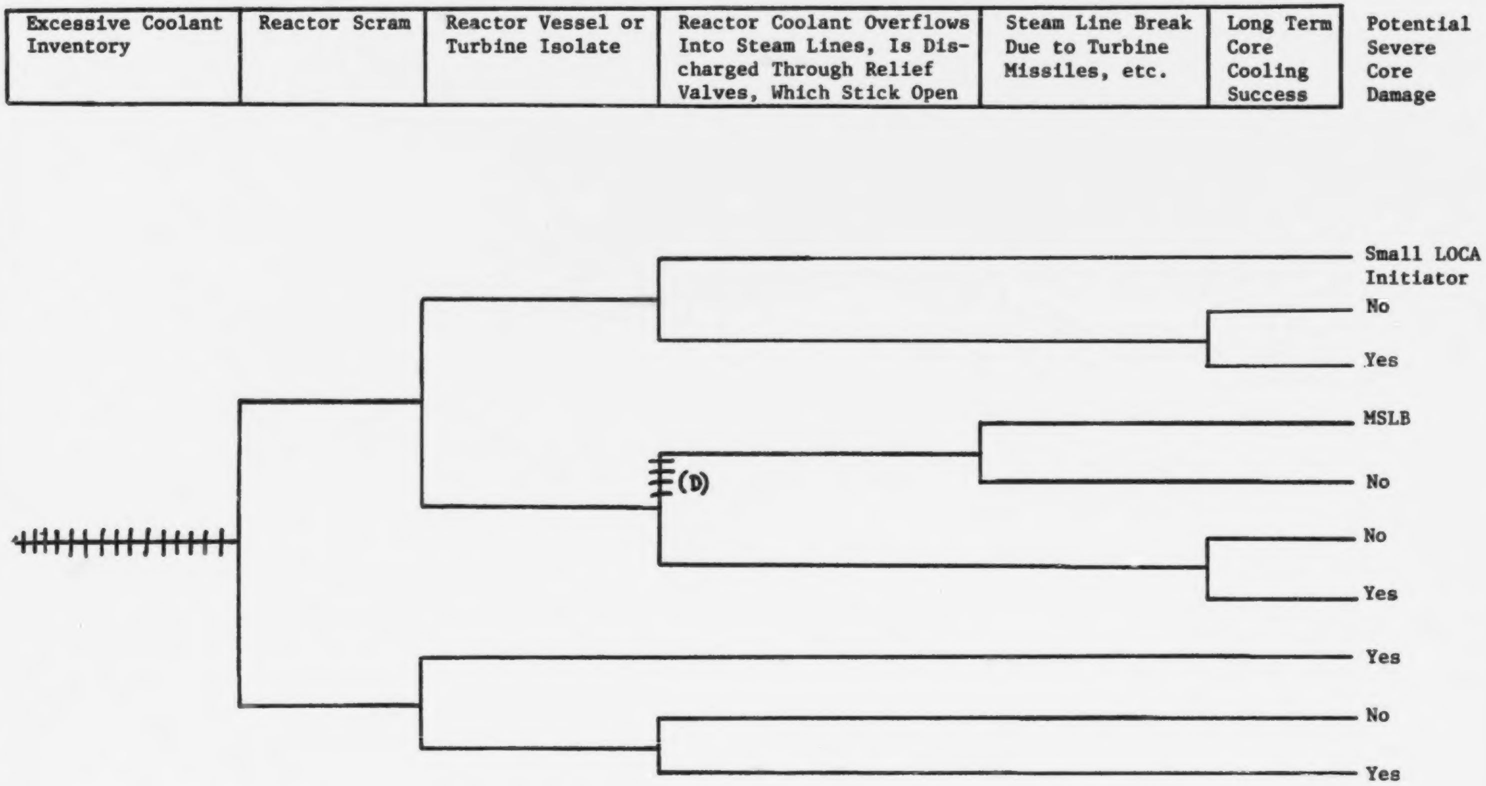
---

[*] Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

BEST DOCUMENT AVAILABLE


A Turbine/Reactor Trip Occurred During Surveillance Testing of the High/Low Level Scram Sensors
The Feedwater System Overfeed The Reactor to Compensate for the Shrinking Water Level
The Operator Incorrectly Took Conrol of the Feedwater System and Caused Water to Flow Into the Steam Lines
Potential Severe Core Damge
MSLB initiator
No
No

NSIC 74242 — Actual Occurrence of High Coolant Level at Nine Mile Point


Excessive Coolant Inventory
Reactor Scram
Reactor Vessel or Turbine Isolate
Reactor Coolant Overflows Into Steam Lines, Is Discharged Through Relief Valves, Which Stick Open
Steam Line Break Due to Turbine Missiles, etc.
Long Term Core Cooling Success
Potential Severe Core Damage
(D)
Small LOCA Initiator
No
Yes
MSLB
No
No
Yes
Yes
No
Yes

NSIC 74242 — Sequence of Interest for High Coolant Level at Nine Mile Point

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 74242

DATE OF LER: January 20, 1972

DATE OF EVENT: December 31, 1971

SYSTEM INVOLVED: Feedwater

COMPONENT INVOLVED: Controls

CAUSE: Operator Error

SEQUENCE OF INTEREST: Excessive Coolant Inventory Transient

ACTUAL OCCURRENCE: High Coolant Level

REACTOR NAME: Nine Mile Point

DOCKET NUMBER: 50-220

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 620 MWe

REACTOR AGE: 2.4 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: Utility

OPERATORS: Niagara Mohawk Power Company

LOCATION: 8 miles NE of Oswego, NY

DURATION: N/A

PLANT OPERATING CONDITION: Greater than 45% of rated load

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; (c) made inoperable; [✓ (d) incorrect operator response]

DISCOVERY METHOD: operational transient

COMMENT:

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 75074

Date: August 17, 1972

Title: Loss of Load and Iodine Release at La Crosse

The failure sequence was:

1. One of the two paralleled 69/161 kv transformers in the switch yard adjacent to the plant was taken out of service to repair an oil leak.
2. Upon completion of the repairs oil was being added to the transformer at a rate which caused operation of the "sudden pressure relay". This relay had not been disabled prior to repair as specified in the proceedures.
3. The relay protecting the turbine generator also tripped for unknown reasons.
4. House loads were transferred from the unit auxiliary to the reserve transformer resulting in a complete loss of generator load.
5. The diesel generator started and assumed safety related loads.

Corrective action:

Maintenance personnel were instructed on the function of the "sudden pressure" relay function and details were incorporated into the transformer maintenance proceedures to prevent inadvertant operations of protective devices.

Design purpose of failed system or component:

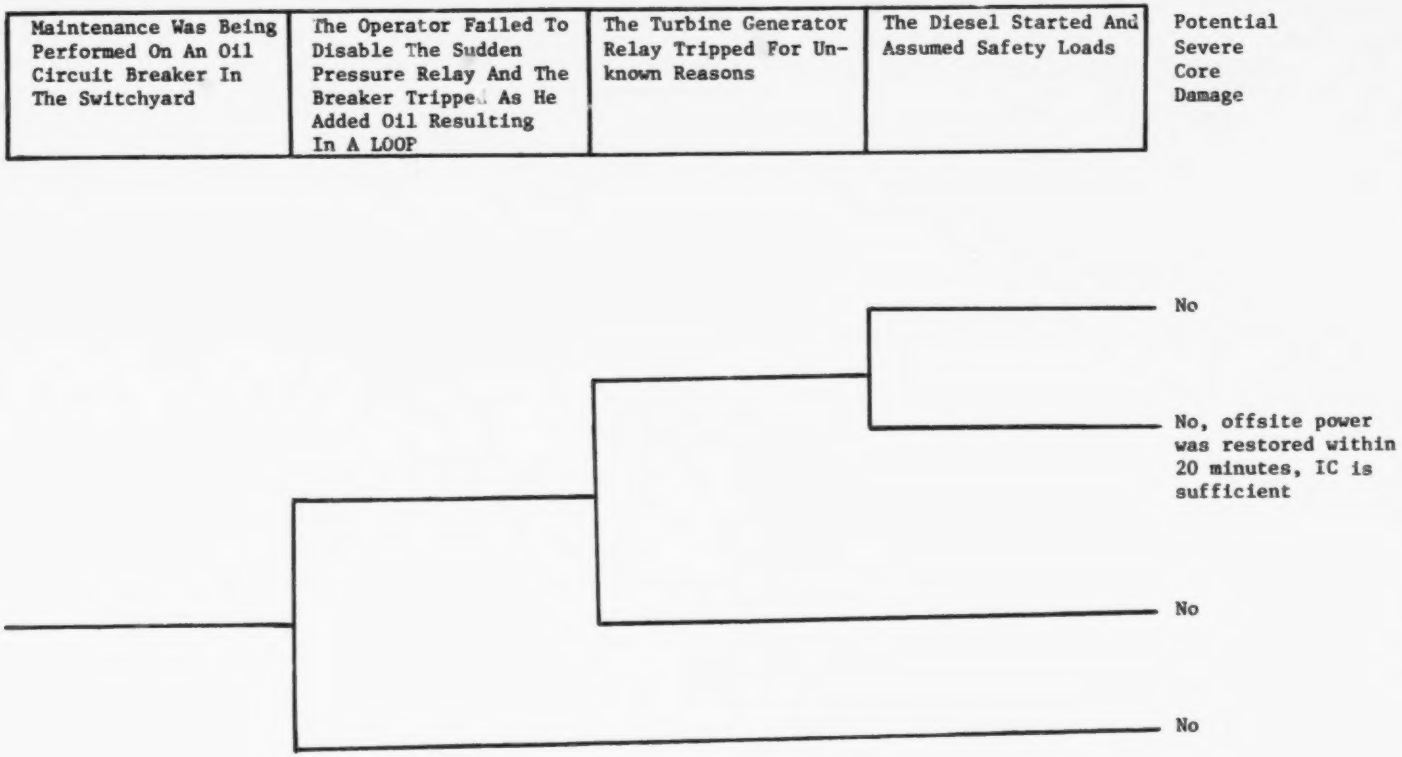
Offsite power provides power to house loads when the generator is down.

Unavailability of system per WASH 1400:[*] LOOP: 2×10^{-5}/hr

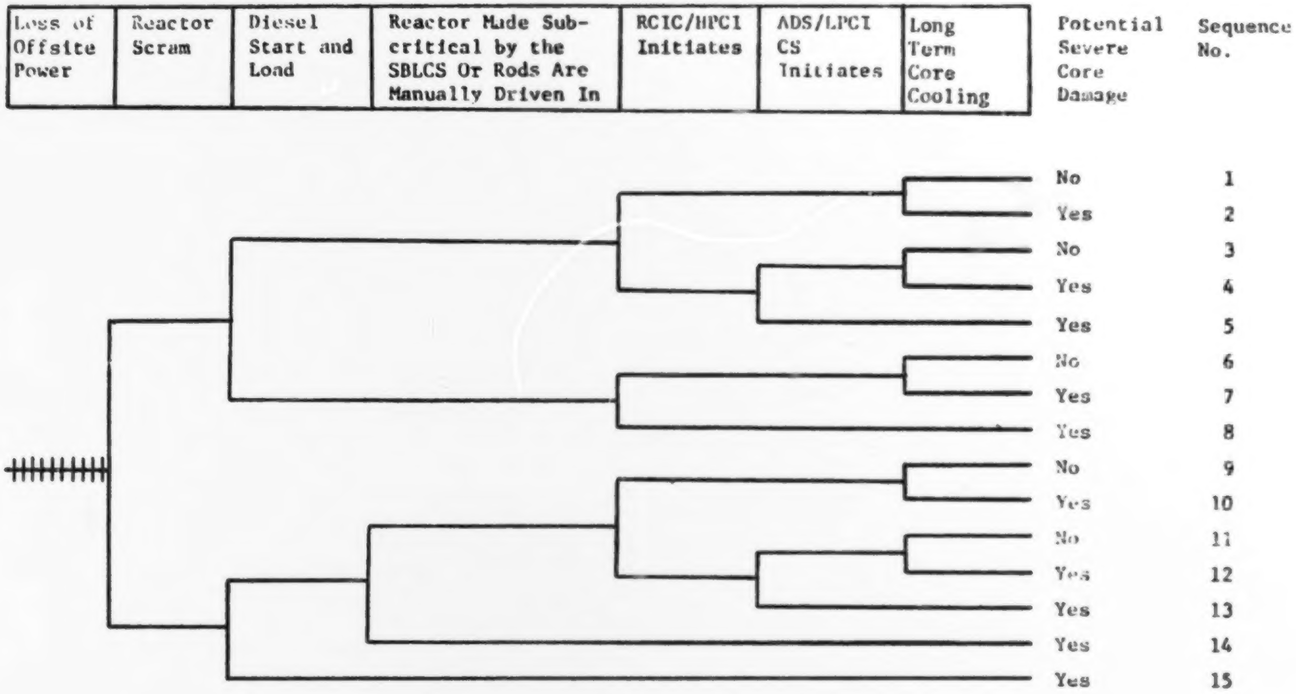
Unavailability of component per WASH 1400:[*] -

---

[*] Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Maintenance Was Being Performed On An Oil Circuit Breaker In The Switchyard
The Operator Failed To Disable The Sudden Pressure Relay And The Breaker Tripped As He Added Oil Resulting In A LOOP
The Turbine Generator Relay Tripped For Unknown Reasons
The Diesel Started And Assumed Safety Loads
Potential Severe Core Damage
No
No, offsite power was restored within 20 minutes, IC is sufficient
No
No

NSIC 75074 — Actual Occurrence of Loss of Load and Iodine Release at LaCrosse


Loss of Offsite Power
Reactor Scram
Diesel Start and Load
Reactor Made Subcritical by the SBLCS Or Rods Are Manually Driven In
RCIC/HPCI Initiates
ADS/LPCI CS Initiates
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
Yes 2
No 3
Yes 4
Yes 5
No 6
Yes 7
Yes 8
No 9
Yes 10
No 11
Yes 12
Yes 13
Yes 14
Yes 15

NSIC 75074 — Sequence of Interest for Loss of Load and Iodine Release at LaCrosse

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 75074

DATE OF LER: September 13, 1972

DATE OF EVENT: August 17, 1972

SYSTEM INVOLVED: Offsite power

COMPONENT INVOLVED: Relays

CAUSE: Operator Error

SEQUENCE OF INTEREST: Loss of Offsite Power

ACTUAL OCCURRENCE: Loss of Load and Iodine Release

REACTOR NAME: La Crosse

DOCKET NUMBER: 407

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 50 MWe

REACTOR AGE: 5.1 yr

VENDOR: Allis Chalmers

ARCHITECT-ENGINEERS: Sargent and Lundy

OPERATORS: Dairyland Power Corporation

LOCATION: 19 miles S of La Crosse

DURATION: N/A

PLANT OPERATING CONDITION: Just scrammed from power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; (c) made inoperable; [✓ (d)] initiating event

DISCOVERY METHOD: Operational Event

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: **77916**

Date: **December 29, 1972**

Title: **Several Valves Malfunction at Oyster Creek**

The failure sequence was:

1. **An operator error occurred resulting in a turbine trip/reactor trip.**
2. **The electromatic relief valve lifted as expected.**
3. **The "D" electromatic relief valve stuck open resulting in a blowdown of approximately 50,000 gal of water.**
4. **The reactor mode switch broke, preventing the operator from switching from "run" to "start up." Thus as the pressure dropped the MSIVs should have closed at low steamline pressure.**
5. **One MSIV failed to automatically close.**
6. **The MSIV closed after manually cycling the valve.**

(see attached page)

Corrective action:

1. **The failure of the electromatic relief valve was traced to the assembly that prevents the retainer ring from rotating. To prevent recurrence of this failure a new method was introduced to inhibit retainer ring rotation.**
2. **The reactor mode switch was replaced.**
3. **A malfunctioning control switch caused the MSIV to fail. This switch was replaced. Plans were made to develop a new MSIV testing schedule and procedure.**

(see attached page)

Design purpose of failed system or component:

1. **Electromatic relief valves provide pressure control in the reactor pressure vessel.**
2. **The MSIV allows the reactor pressure vessel to be isolated.**
3. **The isolation condensers provide an alternate heat sink.**
4. **The vacuum breakers act to prevent the primary containment from exceeding pressure constraints and prevent back flow of water from the suppression pool into the drywell.**

Unavailability of system per WASH 1400:* **vapor suppression pool: 1.2×10^{-3}**

Unavailability of component per WASH 1400:* **check valve: 1×10^{-5}/D, MSIV and isolation valve: 1×10^{-3}/D**

---

[*] Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

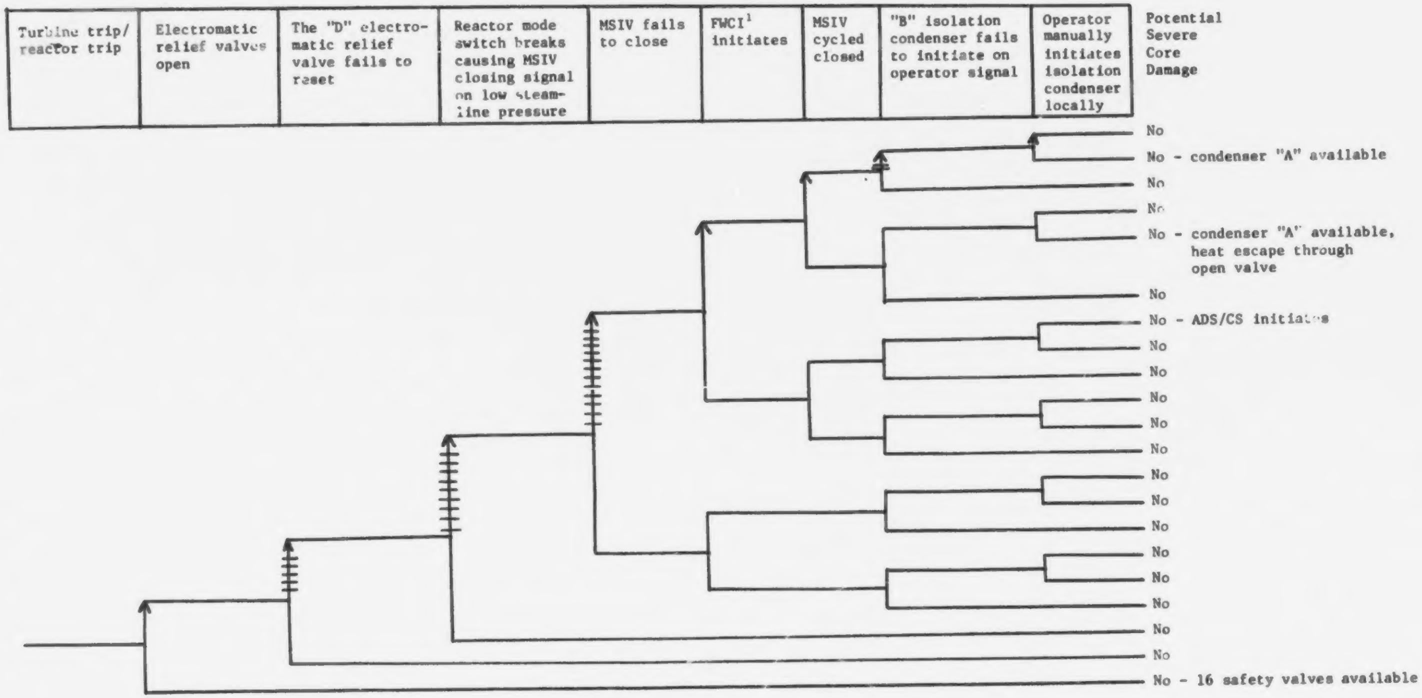
[redacted]

The failure sequence was: (continued)

7. The return valve on the "B: isolation condenser failed to open even though the valve switch was in the open position.
8. The valve was finally opened manually.

Correction action: (continued)

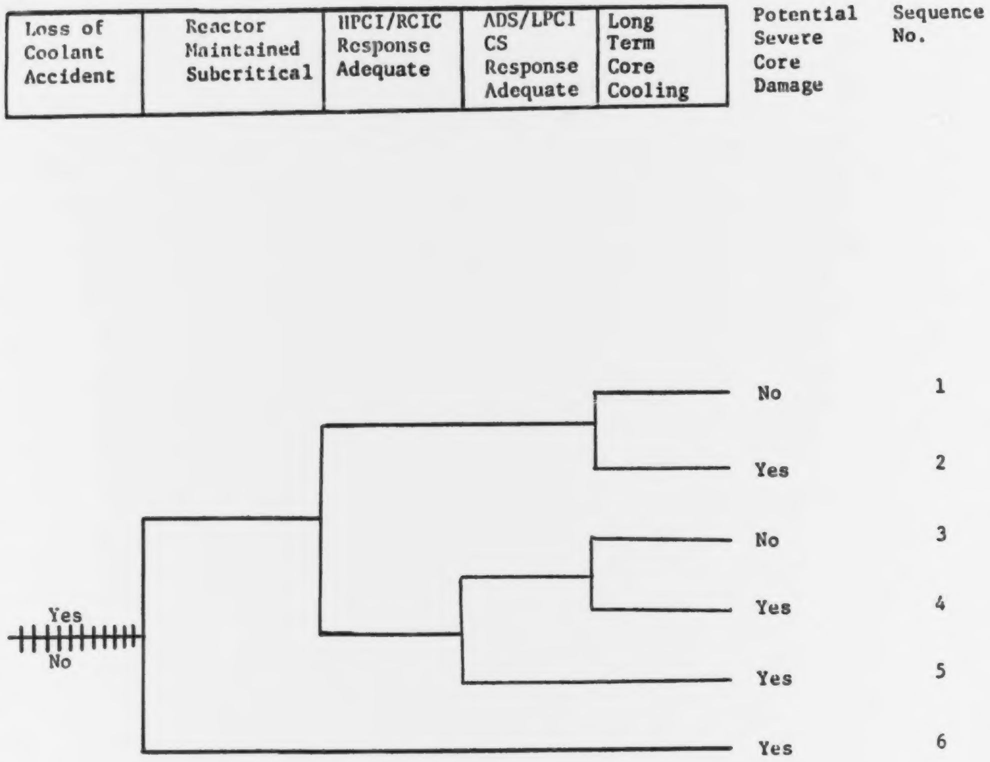
4. The "B" isolation condenser valve was "frozen" in the closed position which caused the valve motor to fail. The valve was removed, repaired, reinstalled and tested satisfactorily.
5. All safety related valves were tested and a vacuum breaker in the suppression pool was found to have failed and another found stuck. Both valves were repaired and tested satisfactorily.


Turbine trip/ reactor trip
Electromatic relief valves open
The "D" electromatic relief valve fails to reset
Reactor mode switch breaks causing MSIV closing signal on low steam-line pressure
MSIV fails to close
FWCI[1] initiates
MSIV cycled closed
"B" isolation condenser fails to initiate on operator signal
Operator manually initiates isolation condenser locally
Potential Severe Core Damage
No
No - condenser "A" available
No
No
No - condenser "A" available, heat escape through open valve
No
No - ADS/CS initiates
No
No
No
No
No
No
No
No
No
No
No
No
No
No - 16 safety valves available

NSIC 77916 — Actual Occurrence of Several Valve Malfunctions at Oyster Creek

[1] FWCI system was presumed to operate.

[2] Only one MSIV failed, thus the core is isolated.


Loss of Coolant Accident
Reactor Maintained Subcritical
HPCI/RCIC Response Adequate
ADS/LPCI CS Response Adequate
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
No
1
Yes
2
No
3
Yes
4
Yes
5
Yes
6

NSIC 77916 — Sequence of Interest for Several Valves Malfunctions at Oyster Creek

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: **77916**

DATE OF LER: **January 17, 1973**

DATE OF EVENT: **December 29, 1972**

SYSTEM INVOLVED: **reactor control, reactor coolant, isolation condensers**

COMPONENT INVOLVED: **valves and switches**

CAUSE: **component failures** operator error

SEQUENCE OF INTEREST: **loss of coolant**

ACTUAL OCCURRENCE: **malfunction of several valves**

REACTOR NAME: **Oyster Creek**

DOCKET NUMBER: **50-219**

REACTOR TYPE: **BWR**

DESIGN ELECTRICAL RATING: **650** MWe

REACTOR AGE: **3.65** yr

VENDOR: **GE**

ARCHITECT-ENGINEERS: **Burns and Roe**

OPERATORS: **Jersey Central Power and Light**

LOCATION: **9 miles S of Toms River, New Jersey**

DURATION: **N/A**

PLANT OPERATING CONDITION: **scrammed**

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ______________________

DISCOVERY METHOD: **operational event**

COMMENT:

BEST DOCUMENT AVAILABLE

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: **78418**

Date: **February 2, 1973**

Title: **Failure of Charging Pump Valves to Open After an Ice Storm at Maine Yankee**

The failure sequence was:

1. **During ECCS valve surveillance testing, the suction valve from the RWST to charging pump 14A failed to open due to ice on the valve stem, a result of an ice storm earlier in the day.**
2. **The suction valve from the RWST to charging pump 14B failed to open completely, also because of ice on the valve stem.**

Corrective action:

The ice was removed from the valve stems and protective covers fabricated for the valves.

Design purpose of failed system or component:

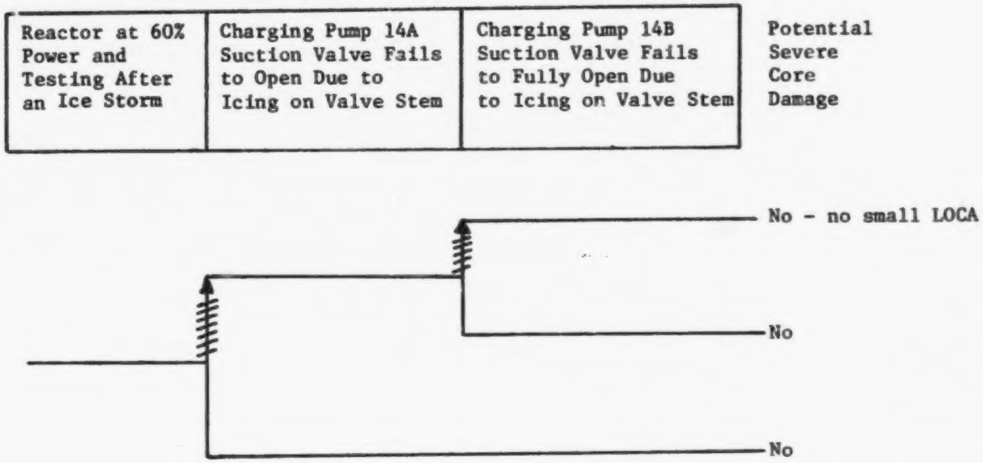
The charging pump suction valves admit water from the RWST to the charging pumps during safeguards actuation, when pump suction flow is automatically switched from the volume control tank to the RWST.

Unavailability of system per WASH 1400:* **High Pressure Injection System (Two of Three Pumps Required on Surry): 1.2×10^{-2}**

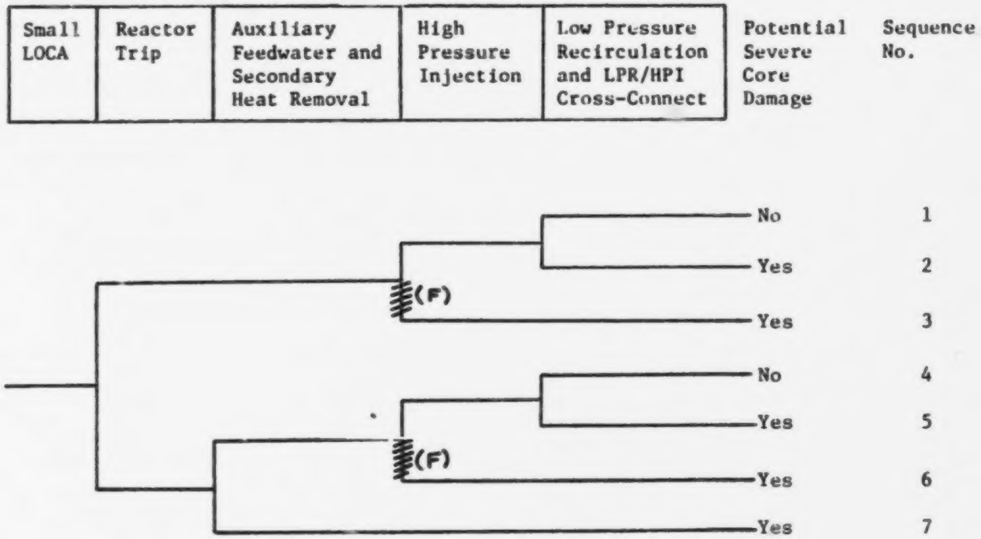
Unavailability of component per WASH 1400:* **Motor Operated Valve, failure to operate: 1×10^{-3}/D**

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor at 60% Power and Testing After an Ice Storm
Charging Pump 14A Suction Valve Fails to Open Due to Icing on Valve Stem
Charging Pump 14B Suction Valve Fails to Fully Open Due to Icing on Valve Stem
Potential Severe Core Damage
No - no small LOCA
No
No

NSIC 48418 — Actual Occurrence for Failure of Charging Pump Valves to Open After an Ice Storm at Maine Yankee


Small LOCA
Reactor Trip
Auxiliary Feedwater and Secondary Heat Removal
High Pressure Injection
Low Pressure Recirculation and LPR/HPI Cross-Connect
Potential Severe Core Damage
Sequence No.
(F)
(F)
No
1
Yes
2
Yes
3
No
4
Yes
5
Yes
6
Yes
7

NSIC 78418 — Sequence of Interest for Failure of Charging Pump Valves to Open After an Ice Storm at Maine Yankee

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 78418

DATE OF LER: February 7, 1973

DATE OF EVENT: February 2, 1973

SYSTEM INVOLVED: High Pressure Injection System

COMPONENT INVOLVED: Charging pump suction valves

CAUSE: Ice on valve stems

SEQUENCE OF INTEREST: small LOCA

ACTUAL OCCURRENCE: valve failure to open during testing

REACTOR NAME: Maine Yankee

DOCKET NUMBER: 50-309

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 825 MWe

REACTOR AGE: .3 yr

VENDOR: Combustion Engineering

ARCHITECT-ENGINEERS: Stone and Webster

OPERATORS: Maine Yankee Atomic Power Co.

LOCATION: 10 miles north of Bath, Maine

DURATION: 360(a) hours

PLANT OPERATING CONDITION: at 60% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ______________________

DISCOVERY METHOD: during testing

COMMENT: –

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: **79565**

Date: **December, 1972**

Title: **Loss of Normal Station Power Causes Cooling System Transient at Vermont Yankee (Operations Report for December, 1972)**

The failure sequence was:

1. **An internal fault on the startup transformer resulted in a loss of offsite power.**
2. **This resulted in a reactor trip followed by a turbine trip.**
3. **The diesel generator started automatically and loaded essential busses.**
4. **The HPCI failed to autostart on high dry well pressure, but was manually initiated.**
5. **The reactor pressure rose to 1130 which resulted in the opening of 3 relief valves. One relief valve failed to open. One safety valve chattered on its seat about 100 psig. below its setpoint.**
6. **RCIC/HPCI were used for decay heat removal and water level control.**
7. **No reason was given for the failure of HPCI to autostart. Failure of the fourth relief valve to lift was attributed to expansion of the valve's components.**

Corrective action:

1. **The chattering safety valve was replaced.**
2. **The failed relief valve components were replaced.**

Design purpose of failed system or component:

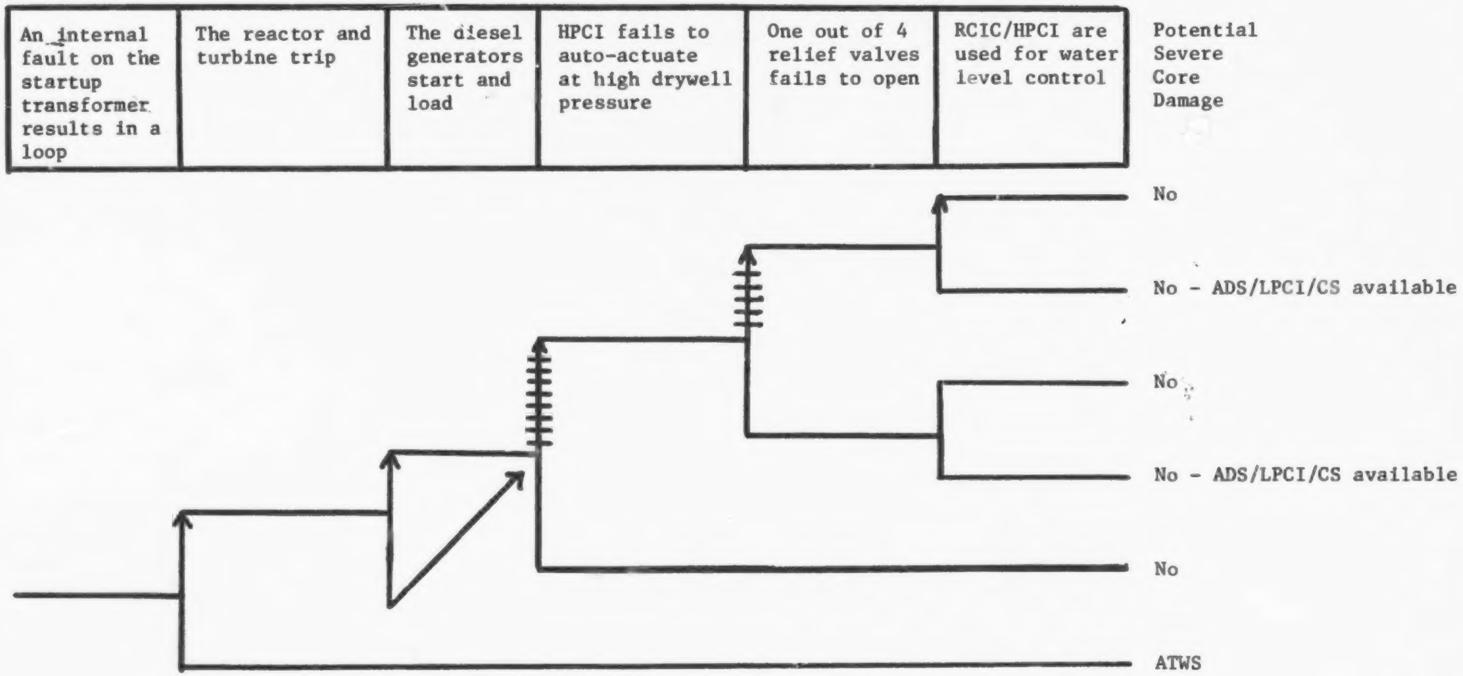
1. **Offsite power provides for station load when the unit is not generating electricity.**
2. **HPCI provides for high pressure injection during a small LOCA.**
3. **The relief valves are used for pressure control.**

Unavailability of system per WASH 1400:* **HPCI: 8.0×10^{-2}/D**
RCIC: 8.0×10^{-2}/D

Unavailability of component per WASH 1400:* **relief valve, failure to open: 1×10^{-5}/D**

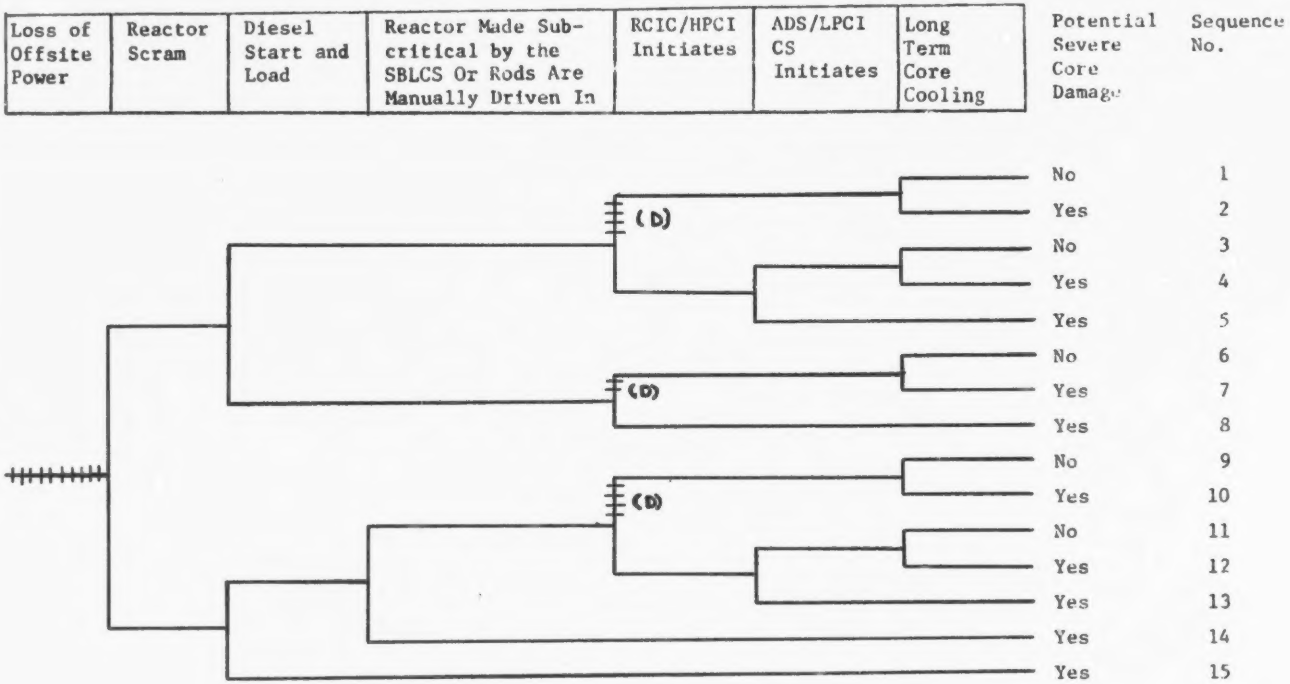
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


An internal fault on the startup transformer results in a loop
The reactor and turbine trip
The diesel generators start and load
HPCI fails to auto-actuate at high drywell pressure
One out of 4 relief valves fails to open
RCIC/HPCI are used for water level control
Potential Severe Core Damage
No
No - ADS/LPCI/CS available
No
No - ADS/LPCI/CS available
No
ATWS

NSIC 79565 — Actual Occurrence of Loss of Normal Station Power Causes Cooling System Transient at Vermont Yankee

BEST DOCUMENT AVAILABLE


Loss of Offsite Power
Reactor Scram
Diesel Start and Load
Reactor Made Sub-critical by the SBLCS Or Rods Are Manually Driven In
RCIC/HPCI Initiates
ADS/LPCI CS Initiates
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
(D)
(D)
(D)
No 1
Yes 2
No 3
Yes 4
Yes 5
No 6
Yes 7
Yes 8
No 9
Yes 10
No 11
Yes 12
Yes 13
Yes 14
Yes 15

NSIC 79565 — Sequence of Interest of Loss of Normal Station Power Causes Cooling System Transient at Vermont Yankee

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 79565

DATE OF LER: December 1, 1972

DATE OF EVENT: December, 1972

SYSTEM INVOLVED: HPCI, ADS

COMPONENT INVOLVED: relief valve

CAUSE: Unspecified for HPCI, mechanical failure of valve components

SEQUENCE OF INTEREST: Loss of Offsite Power

ACTUAL OCCURRENCE: Loss of Normal Station Power

REACTOR NAME: Vermont Yankee

DOCKET NUMBER: 50-271

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 514 MWe

REACTOR AGE: .8 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: Ebasco

OPERATORS: Vermont Yankee Nuclear Power Corp.

LOCATION: 5 miles S of Battleboro, VT

DURATION: N/A

PLANT OPERATING CONDITION: 63% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________________

DISCOVERY METHOD: Operational Event

COMMENT: —

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 80138

Date: June 10, 1972

Title: Valve Problem Causes Flooding of Turbine-Building Basement at Quad Cities 1

The failure sequence was:

1. The 10' diameter butterfly valves in the circulating water system of the main condenser were being modified during a scheduled maintenance outage. Operators were venting the actuater on a valve in one of the circulating water discharge lines. The valve slammed shut. Three circulating water pumps were operating when the valve slammed shut and this resulted in a water hammer which ruptured a rubber expansion joint in the recirculation line.
2. The flooding was immediately reported to the control room and operators shut down the pumps. However during the interim (∿6 min) river water flooded the turbine building basement to a depth of ∿15.5 ft.
3. Several pieces of equipment were flooded including the following safety related equipment:
   1. four emergency service water pumps (RHR system cooling)
   2. cooling water pumps for the unit diesel (#1) and the swing diesel (#1/2).

Corrective action:

1. The water was pumped out of the basement and was continuously monitored for radiation.
2. The damaged equipment was repaired.
3. New procedures were adopted for venting the valve hydraulic actuation.
4. Flood alarms were installed, and a "very high level" trip was installed to shut down the circulating water pump should an expansion joint rupture while the pumps are in operation.

(see attached sheet)

Design purpose of failed system or component:

1. The emergency service water pumps provide cooling water to the RHR heat exchangers.
2. The diesels provide emergency power should all offsite power and unit power be lost.

Unavailability of system per WASH 1400:* RHR (HPSW): $4 \times 1 \times 10^{-5}$/D

Unavailability of component per WASH 1400:* --

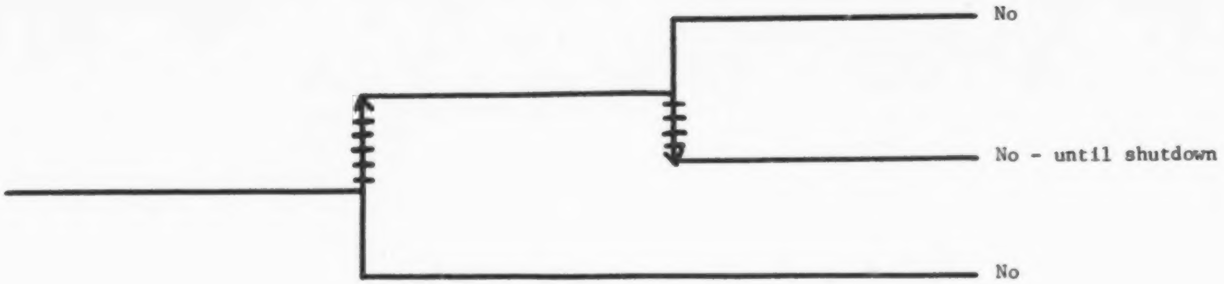
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

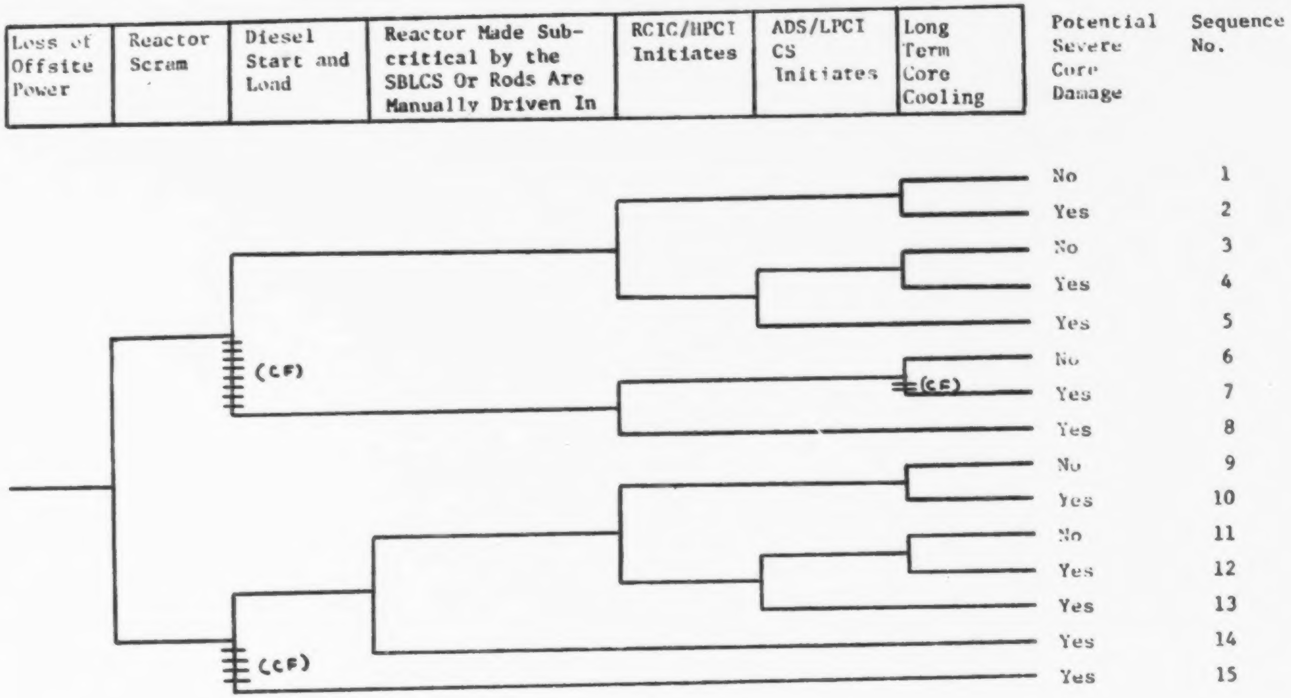
Corrective action: (continued)

5. Safety related equipment located in the condenser pump room was enclosed in water tight vaults.
6. The ability to cross tie RHR systems of the two units was also installed.

10' butterfly valve slams shut as operator vents valve actuator. A water hammer results from the valve slamming and the operation of 3 circulating water pumps	The water hammer results in a rupture of 9 snubber expansion joints	Circulating water pumps shutdown in time to prevent loss of safety related equipment	Potential Severe Core Damage



NSIC 80138 — Actual Occurrence of Valve Flooding of the Turbine Building Basement at Quad Cities 1


Loss of Offsite Power
Reactor Scram
Diesel Start and Load
Reactor Made Subcritical by the SBLCS Or Rods Are Manually Driven In
RCIC/HPCI Initiates
ADS/LPCI CS Initiates
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
(CF)
(CF)
(CF)
No 1
Yes 2
No 3
Yes 4
Yes 5
No 6
Yes 7
Yes 8
No 9
Yes 10
No 11
Yes 12
Yes 13
Yes 14
Yes 15

NSIC 80138 – Sequence of Interest for Valve Problem Causes Flooding of Turbine Building Basement at Quad Cities 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 80138

DATE OF LER: June 19, 1972

DATE OF EVENT: June 10, 1972

SYSTEM INVOLVED: RHR, electric power

COMPONENT INVOLVED: All 4 RHR emergency service water pumps, cooling water pumps to diesels 1 and 1/2.

CAUSE: ruptured expansion joint (human error)

SEQUENCE OF INTEREST: loss of offsite power

ACTUAL OCCURRENCE: Valve Problem Causes Flooding of Turbine Building at Quad Cities 1

REACTOR NAME: Quad Cities 1

DOCKET NUMBER: 50-254

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 769 MWe

REACTOR AGE: .69 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: Sargent & Lundy

OPERATORS: Commonwealth Edison

LOCATION: 20 miles NE of Moline, Ill.

DURATION: 5 (a) hours

PLANT OPERATING CONDITION: Shutdown

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________________

DISCOVERY METHOD: Operational event

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 81523

Date: June 28, 1973

Title: Failure of AFW Pumps to Auto-Start at Turkey Point 4

The failure sequence was:

1. During startup and low power physics testing, the turbine generator control valves opened rapidly and excessively (reason unspecified).
2. As a result of high steam flow and reduced reactor coolant system temperature, safety injection actuated.
3. All three auxiliary feed pumps failed to start due to failure to install 125V DC power supply fuses in the AFW pump auto-start logic circuits.
4. The AFW pumps were manually started.

Corrective action:

The fuses were installed and the auto-start logic circuits were demonstrated to be operable.

Design purpose of failed system or component:

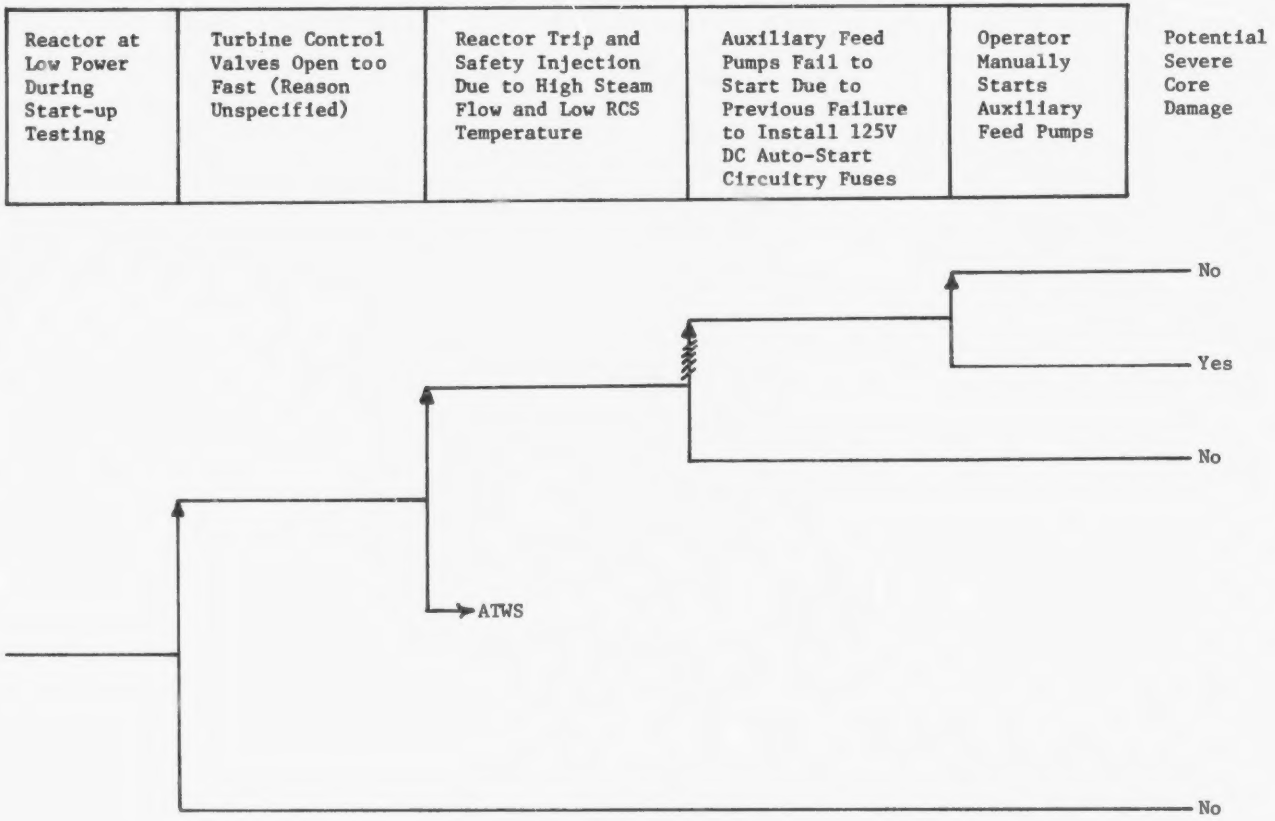
The system provides automatic starting of the AFW pumps on safety injection, steam generator low-low level, loss of voltage on the 4160V busses, and loss of both main feedwater pumps.

Unavailability of system per WASH 1400:* AFW (start + 8 hours): 3.7×10^{-5}

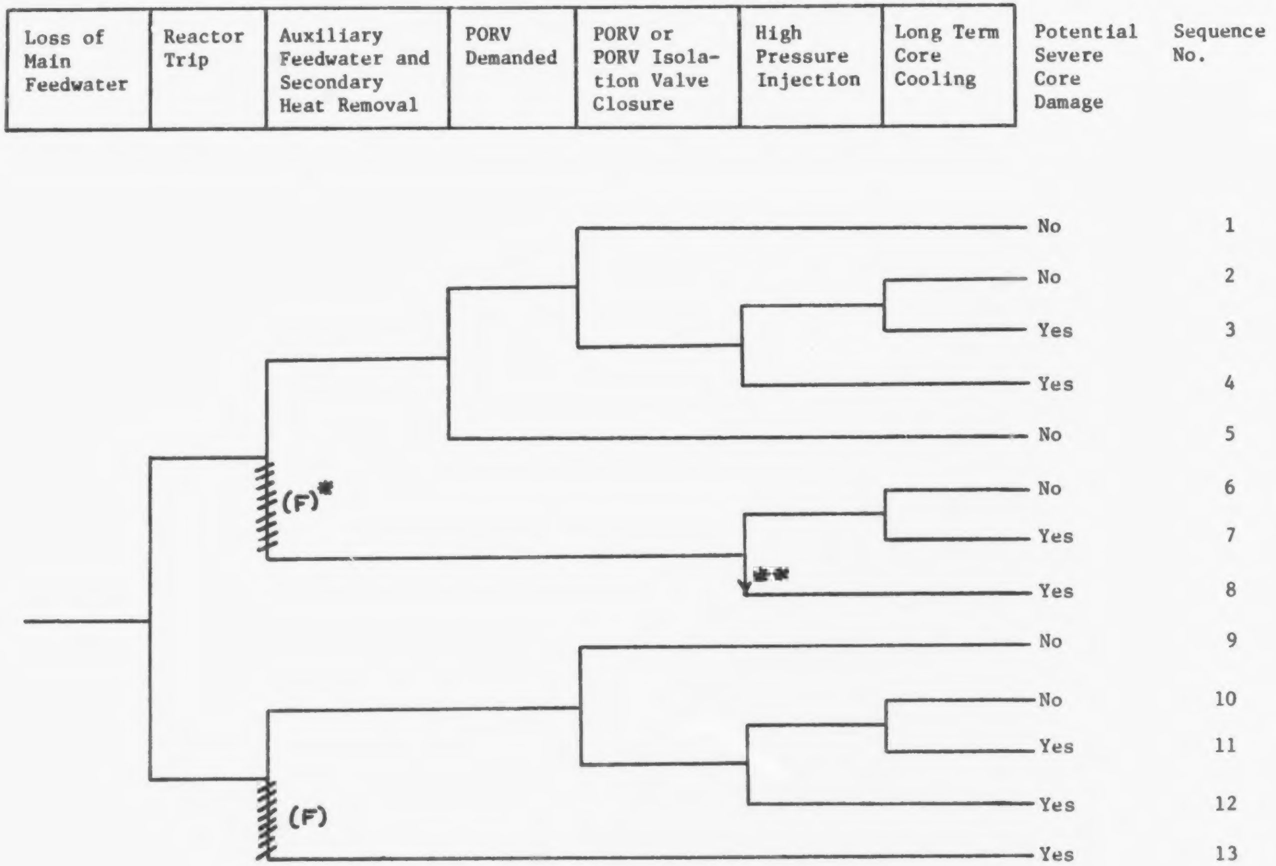
Unavailability of component per WASH 1400:* Fuse: 1×10^{-6}/hr

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor at Low Power During Start-up Testing
Turbine Control Valves Open too Fast (Reason Unspecified)
Reactor Trip and Safety Injection Due to High Steam Flow and Low RCS Temperature
Auxiliary Feed Pumps Fail to Start Due to Previous Failure to Install 125V DC Auto-Start Circuitry Fuses
Operator Manually Starts Auxiliary Feed Pumps
Potential Severe Core Damage
No
Yes
No
ATWS
No

NSIC 81523 - Actual Occurrence for Failure of AFW Pumps to Auto-Start at Turkey Point 4


Loss of Main Feedwater
Reactor Trip
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
(F)*
**
(F)
No 1
No 2
Yes 3
Yes 4
No 5
No 6
Yes 7
Yes 8
No 9
No 10
Yes 11
Yes 12
Yes 13

NSIC 81523 — Sequence of Interest for Failure of AFW Pumps to Auto-Start at Turkey Point 4

*Success requires manual start

**Not included in operating procedures

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 81523

DATE OF LER: June 28, 1973

DATE OF EVENT: June 18, 1973

SYSTEM INVOLVED: Auxiliary feedwater

COMPONENT INVOLVED: Fuses

CAUSE: Failure to install

SEQUENCE OF INTEREST: Loss of main feedwater

ACTUAL OCCURRENCE: Turbine/reactor trip and subsequent loss of main feedwater

REACTOR NAME: Turkey Point 4

DOCKET NUMBER: 50-251

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 693 MWe

REACTOR AGE: .02 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Florida Power and Light

LOCATION: 25 miles South of Miami, Florida

DURATION: 168 hours

PLANT OPERATING CONDITION: At power (low power testing)

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; [✓ (b)] failed to start; (c) made inoperable; (d) ____________________

DISCOVERY METHOD: Operational event

COMMENT:

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 82198

Date: July 13, 1973

Title: Failure to Transfer Auxiliary Power to Startup Transformer at Turkey Point 3

The failure sequence was:
1. With the reactor at 87% power, calibration of steam line pressure transmitters was in progress.
2. Inadvertant safety-injection resulted from coincident trips of two channels of No. 3C steam line high differential pressure. (One channel was in test and another channel received a transient signal to trip.)
3. Following initiation, the reactor tripped and all safety features operated satisfactorily.
4. Automatic transfer from the auxiliary transformer to the startup transformer did not occur due to an unconnected wire in the automatic transfer circuitry.
5. The operator manually closed the startup transformer breakers to the safety-related buses.

Corrective action:
1. The unconnected wire in the automatic transfer circuitry was properly connected and the completed circuit was tested.

Design purpose of failed system or component:

The failed circuit provided for automatic transfer of the safety-related buses to the offsite power source in the event of a loss of power from the unit generator.

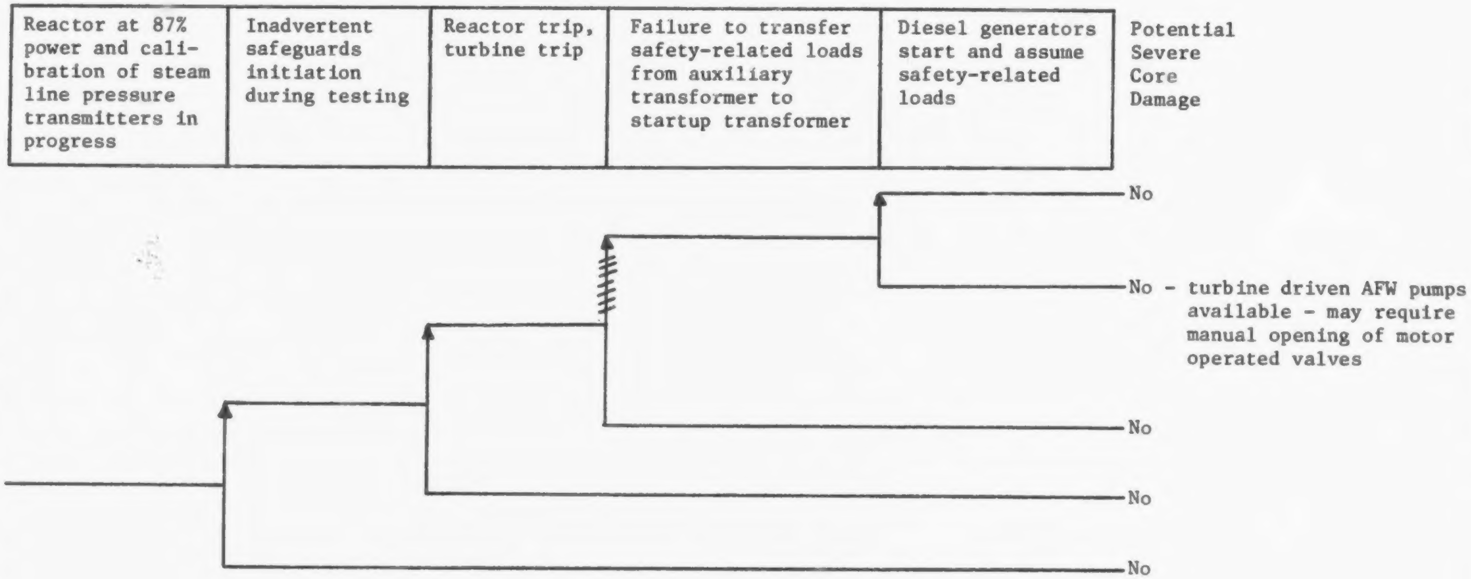
Unavailability of system per WASH 1400:* offsite power: 10^{-3}/D following trip

Unavailability of component per WASH 1400:* --

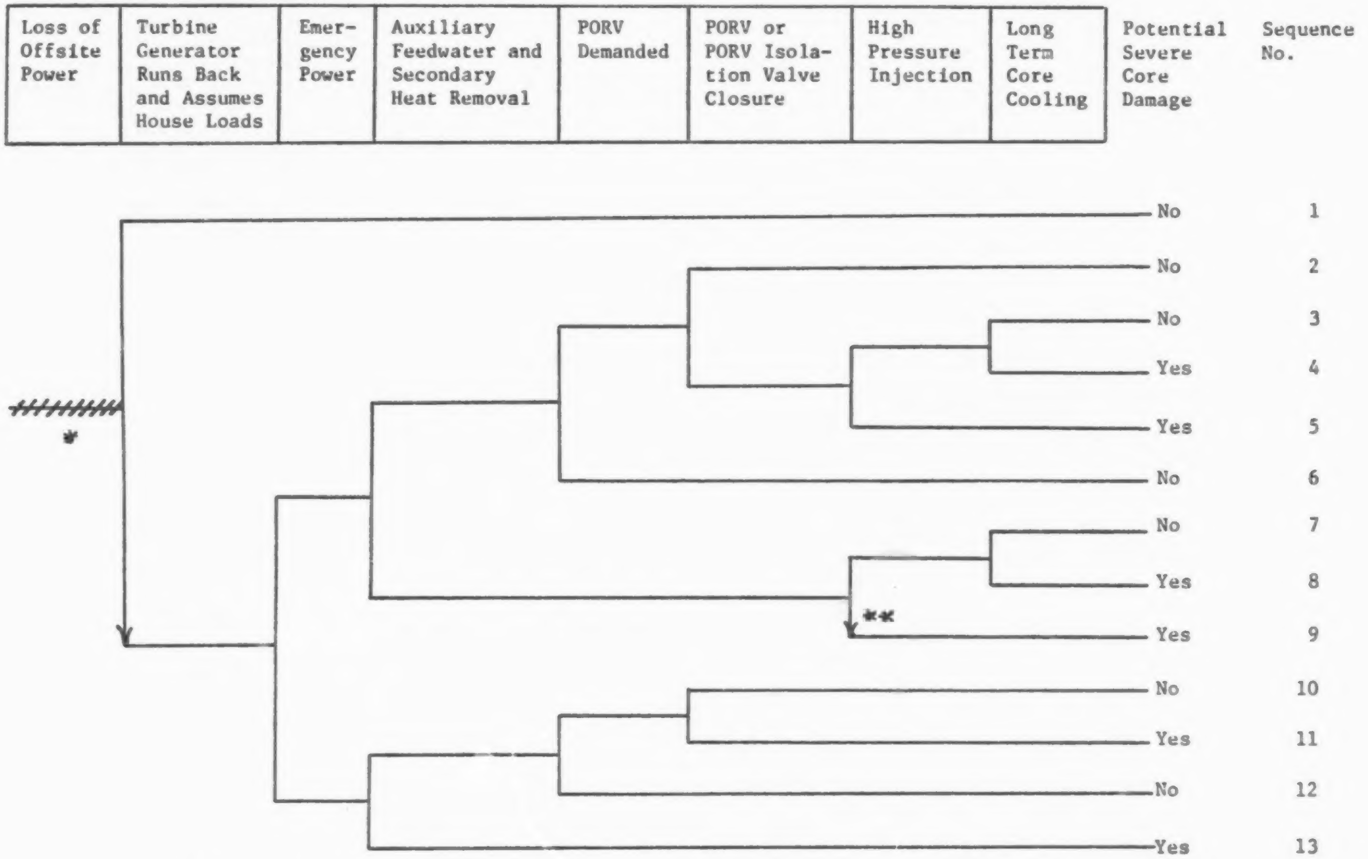
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

BEST DOCUMENT AVAILABLE


Reactor at 87% power and calibration of steam line pressure transmitters in progress
Inadvertent safeguards initiation during testing
Reactor trip, turbine trip
Failure to transfer safety-related loads from auxiliary transformer to startup transformer
Diesel generators start and assume safety-related loads
Potential Severe Core Damage
No
No - turbine driven AFW pumps available - may require manual opening of motor operated valves
No
No
No

NSIC 82198 — Actual Occurrence for Failure to Transfer Auxiliary Power to Startup Transformers at Turkey Point 3


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
*
**
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
Yes 9
No 10
Yes 11
No 12
Yes 13

NSIC 82198 — Sequence of Interest for Failure to Transfer Auxiliary Power to Startup Transformers at Turkey Point 3 (effective loss of offsite power)

*Success requires manual transfer to the startup transformer.

**Not included in mitigation procedures.

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 82198

DATE OF LER: July 13, 1973

DATE OF EVENT: July 4, 1973

SYSTEM INVOLVED: plant electrical system

COMPONENT INVOLVED: auxiliary to startup transformer transfer circuitry

CAUSE: unconnected wire

SEQUENCE OF INTEREST: loss of offsite power

ACTUAL OCCURRENCE: inadvertant reactor and turbine trip and safeguards initiation during testing

REACTOR NAME: Turkey Point 3

DOCKET NUMBER: 50-250

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 693 MWe

REACTOR AGE: .7 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Florida Power and Light Company

LOCATION: 25 mi. south of Miami, Florida

DURATION: N/A

PLANT OPERATING CONDITION: 87% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; (c) made inoperable; [✓ (d)] failed to transfer

DISCOVERY METHOD: operational event

COMMENT: --

BEST DOCUMENT AVAILABLE 130

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: **83833**

Date: **September 8, 1973**

Title: **Total Loss of AC Power at Oyster Creek**

The failure sequence was:

1. A plant shutdown was in progress and had progressed to the point where the electrical output was 90 MWe. When attempting to switch the load from the auxiliary transformer to the startup transformer a loss of power occurred on the 1A 4160 V a.c. bus. This was caused by an improper setting on the C phase differential relay after maintenance. The 1C 4160 V bus which is feed from the 1A 4160 bus or the emergency diesel also deenergized. The loss of these buses caused several pumps to trip, including the condensate pumps.

2. Diesel generator number 1 initiated in the "fast start" mode, re-energizing the IC 4160 bus.

3. An attempt was made to restart the stopped B and C condensate pumps while the unit was still generating electricity but the reactor scrammed on low water level, (see attached sheet)

Corrective action:

1. Improper setting of the current transformer ratio matching taps for the C phase differential relays on both S1A and S1B was responsible for the loss of offsite power. The taps were properly set and the transformer returned to service.

2. The problem associated with the diesels was traced to the design of the diesel logic circuit. General Motors was contacted about this problem. (see attached sheet)

Design purpose of failed system or component:

1. Offsite power supplies power to the loads when the unit generator isn't available.

2. The diesel generators supply power to safety related loads when both the unit generator and the offsite power source are unavailable.

Unavailability of system per WASH 1400:* **offsite power: 2 x 10^{-5}/hr.**

Unavailability of component per WASH 1400:* **--**

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

[redacted]

PRECURSOR DESCRIPTION AND DATA   NSIC Accession Number: 83833

The failure sequence was: (continued)

which caused the 1B 4160 V bus to de-energize.

4. Following the scram:
   a. the 1B 4160 V a.c. was energized from the SB transformer
   b. the lockout relay, 86SA was manually reset
   c. the S1A breaker was closed
   d. diesel generator 1 was synchronized with the system
   e. the 1C breaker was closed
5. Diesel number 1 was secured at which time a trouble light was initiated in the control room.
6. Breaker 51A tripped when the operator attempted to start the A condensate pump. Diesel generator 1 didn't start due to the previous actuation of the engine lock out relay.
7. The operator next attempted to start either B or C condensate pump. This resulted in tripping startup transformer S1B. Diesel generator 2 was initiated in the fast start mode.
8. Both the S1A and the S1B breakers and the diesel engine lockout relays were manually reset and closed and diesel generator 1 was started.
9. Diesel generator #2 was synchronized with the line, the 1D breaker was closed and the diesel was secured.
10. The operator again tried to start the B or C condensate pumps after closing the 1D breaker and securing the diesel. This resulted in a trip of breaker 1D. Diesel 2 failed to start due to lockout relay.
11. By this time the water level had dropped to 9' above the core (1'10" above the core spray initiation level). The operator opened the CRD hydraulic control station by pass valve to allow water from the CRD pumps to be pumped into the reactor to counter what was being lost due to decay heat. He finally isolated the reactor by closing the MSIVS and initiated the isolation condensers to remove decay heat.

Corrective action: (continued)

3. If the operator had initially relied upon the diesels (as he finally ended up doing) instead of trying to tie to the offsite power source, the event would have proceeded only to step 4.

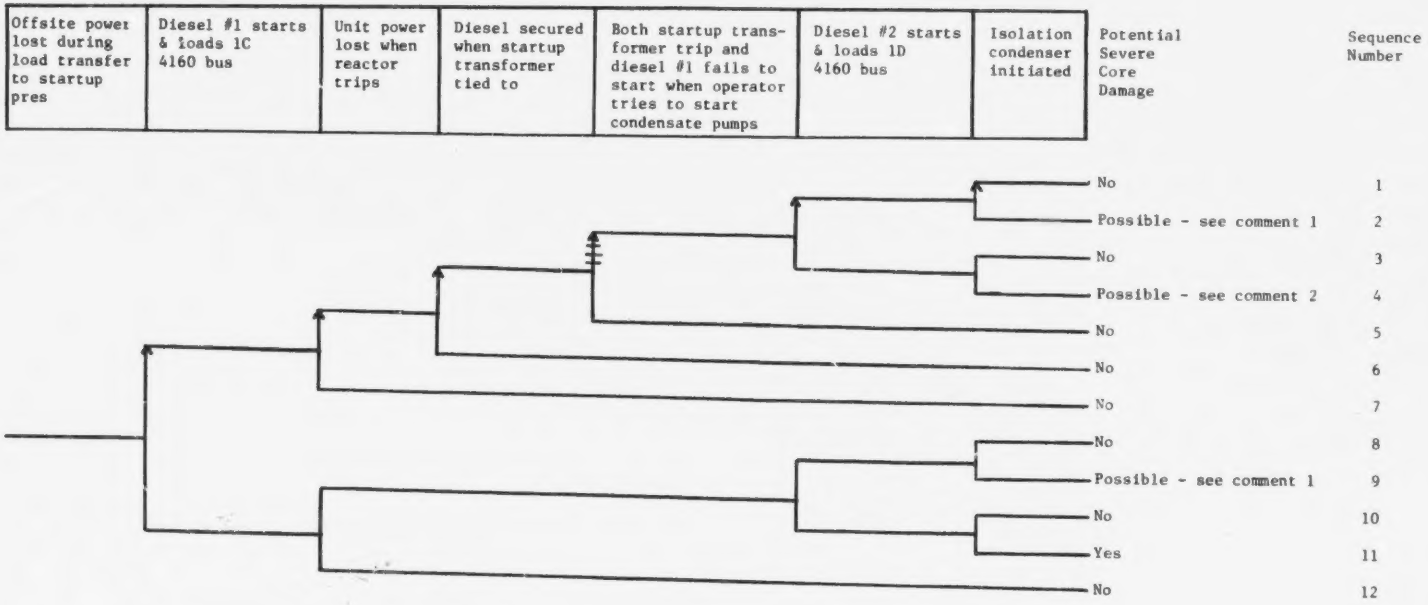
   The following long range steps were taken to prevent a recurrence of this type of event:

   1. Prepare procedures, in conjunction with the Relay Department, for all Relay Department work in the Oyster Creek plant. All work in the future will be coordinated with the shift foreman.

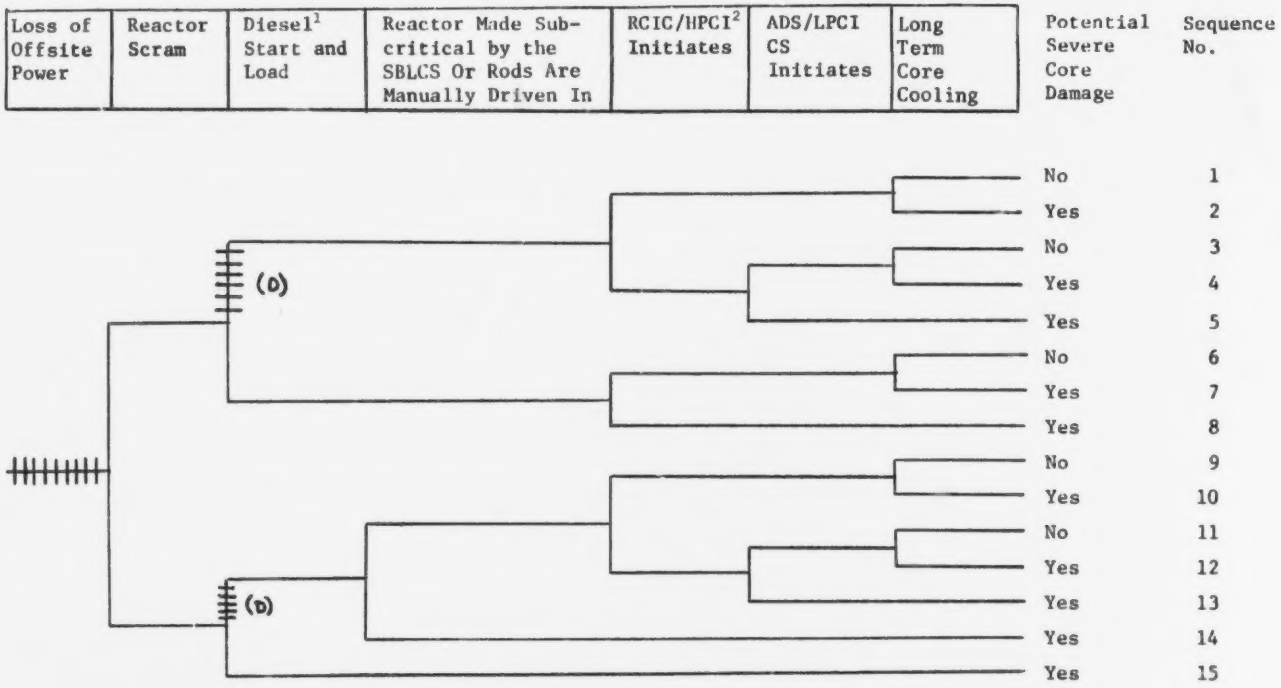
(see attached sheet)

Corrective action: (continued)

2. Prepare a design change to the diesel generator logic circuits to prevent lockout following shutdown of the diesel. General Motors was to be contacted for recommendations.
3. Review and revise Procedure No. 511 "Loss of Feedwater System" as required.


Offsite power lost during load transfer to startup pres
Diesel #1 starts & loads 1C 4160 bus
Unit power lost when reactor trips
Diesel secured when startup transformer tied to
Both startup transformer trip and diesel #1 fails to start when operator tries to start condensate pumps
Diesel #2 starts & loads 1D 4160 bus
Isolation condenser initiated
Potential Severe Core Damage
Sequence Number
No 1
Possible - see comment 1 2
No 3
Possible - see comment 2 4
No 5
No 6
No 7
No 8
Possible - see comment 1 9
No 10
Yes 11
No 12

NSIC 83833 — Actual Occurrence for a Total Loss of A.C. Power at Oyster Creek


Loss of Offsite Power
Reactor Scram
Diesel[1] Start and Load
Reactor Made Subcritical by the SBLCS Or Rods Are Manually Driven In
RCIC/HPCI[2] Initiates
ADS/LPCI CS Initiates
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
(D)
(D)
No 1
Yes 2
No 3
Yes 4
Yes 5
No 6
Yes 7
Yes 8
No 9
Yes 10
No 11
Yes 12
Yes 13
Yes 14
Yes 15

NSIC 83833 — Sequence of Interest for a Total Loss of A.C. Power at Oyster Creek

---

[1] Diesel had initiated earlier in the transient but was locked out after being secured. The total loss of a.c. power lost approximately 15 sec.

[2] Oyster Creek utilizes isolation condenser rather than RCIC.

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 83833

DATE OF LER: September 18, 1973

DATE OF EVENT: September 8, 1973

SYSTEM INVOLVED: electric power

COMPONENT INVOLVED: transformer relays, diesel generator relays

CAUSE: maintenance fault, error in diesel start logic, human error

SEQUENCE OF INTEREST: loss of offsite power

ACTUAL OCCURRENCE: loss of offsite power

REACTOR NAME: Oyster Creek

DOCKET NUMBER: 50-219

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 650 MWe

REACTOR AGE: 4.29 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: Burns and Roe

OPERATORS: Jersey Central Power and Light

LOCATION: 9 miles from Toms River, New Jersey

DURATION: N/A

PLANT OPERATING CONDITION: 90 MWe and proceeding to shutdown

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c) made inoperable]; (d) ______________________

DISCOVERY METHOD: operations

COMMENT: "The designed redundancy for the station vital power supplies (i.e., offsite power and diesel generators) was not present and in fact, had not been present since July 30, 1973..." This is about a 5 week period.

This statement was taken from the LER.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 85370

Date: October 31, 1973

Title: Loss of Offsite Power, Excessive RCS Cooldown, Subsequent Safety Injection, and Failure of a Vital Instrument Bus at R. E. Ginna

The failure sequence was:

1. The reactor was at approximately 100% power, with one of four transmission circuits out of service due to construction and auxiliary power provided from the station switchyard and an external 34.5kv line.
2. A second of four 115kv circuits tripped due to a ground fault.
3. Power swings because of high generated power with only two transmission lines cause remaining two 115kv transmission lines to trip, which resulted in loss of the station switchyard and one of two auxiliary power sources.
4. Loss of all 115kv transmission lines overload a supply circuit for the remaining auxiliary power source, causing it to trip, resulting in a total loss of offsite power, and a turbine trip.

(See attached page)

Corrective action:

1. Operating procedures were revised to reduce the power output from Ginna when any transmission circuit is out of service.
2. The turbine-driven AFW pump control logic was being reviewed because of the rapid steam generator filling.
3. No cause for the momentary instrument bus failure was found.

Design purpose of failed system or component:

1. Offsite electrical power provides an alternate source of power to plant auxiliaries in the event the unit generator is inoperable.
2. The vital instrument bus provides battery-backed uninterruptible power for safety-related instrument loads.

Unavailability of system per WASH 1400:* Offsite power: 2×10^{-5}/hr
Vital instrument bus: 1.7×10^{-5}/hr

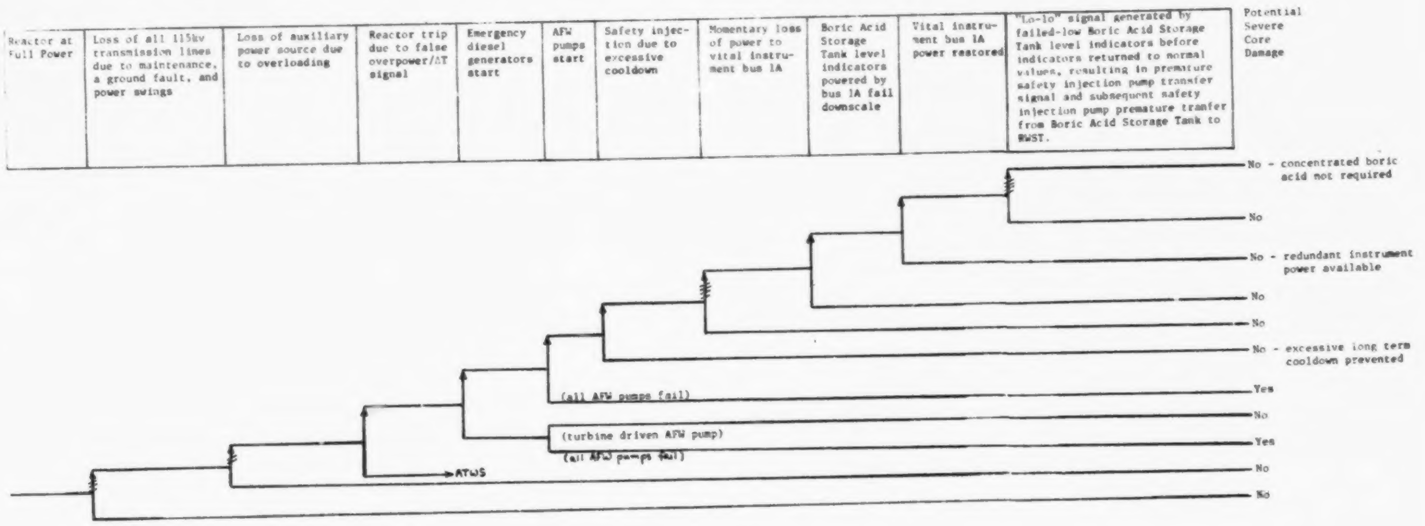
Unavailability of component per WASH 1400:*

---

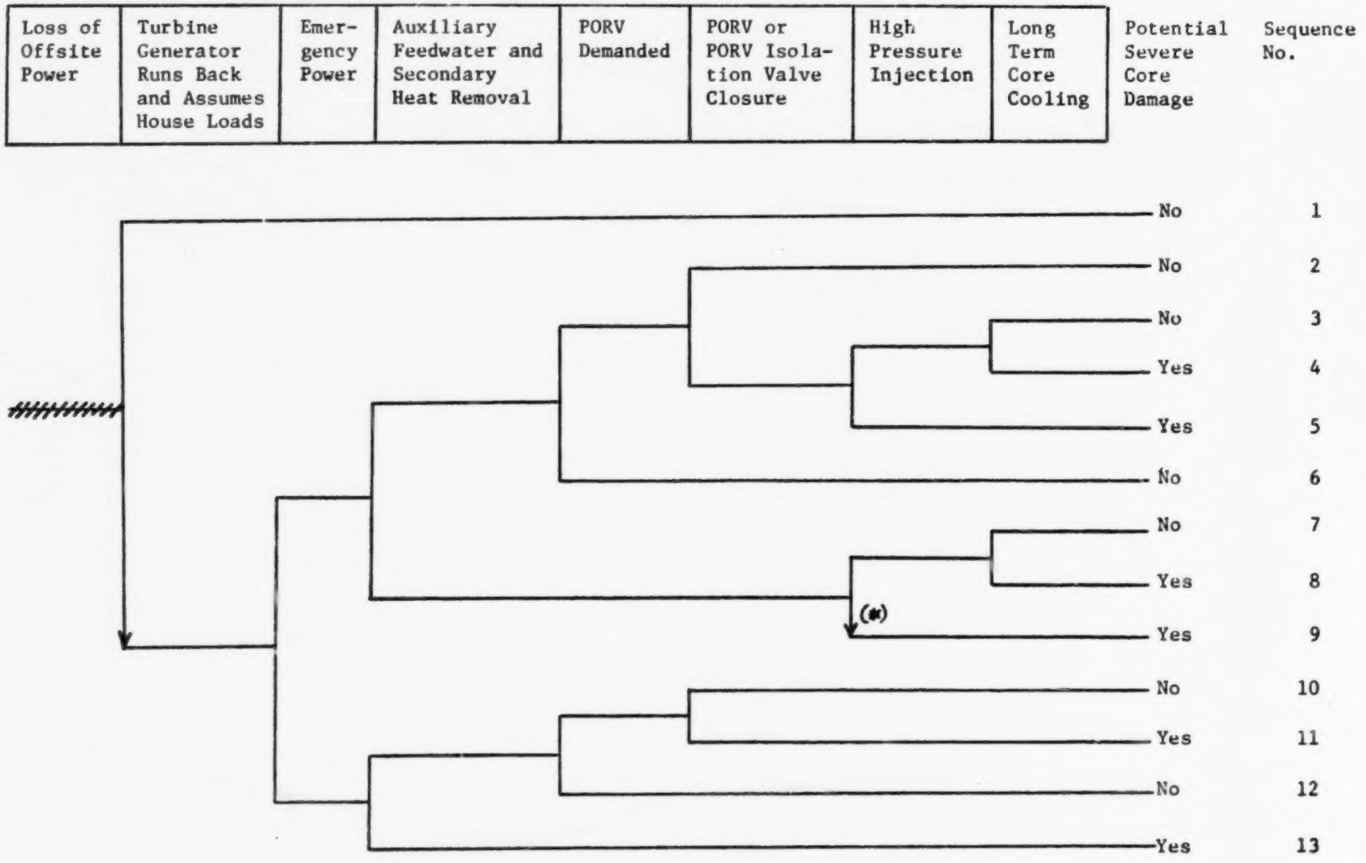
*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

The failure sequence was: (Continued)

5. An electrical disturbance on an instrument bus causes a reactor trip on a false overpower/high ΔT signal.
6. The emergency diesel generators started.
7. The motor-driven and turbine-driven auxiliary feed pumps started on low steam generator level and loss of offsite power.
8. The operator secured the motor-driven AFW pump from the control room and the turbine-driven AFW pump locally because of increasing steam generator water level and decreasing RCS temperature. Because of the turbine-driven pump control circuitry during a loss of offsite power, the pump could only be secured locally.
9. Safety injection was automatically initiated due to low pressurizer pressure and level resulting from the RCS cooldown.
10. Vital bus 1A momentarily failed and caused the Boric Acid Storage Tank level transmitters powered from this bus to fail downscale.
11. Vital bus 1A power was restored.
12. The failed-low Boric Acid Storage Tank level indicators generated a "lo-lo" level signal before returning to their normal position, resulting in a premature safety injection pump transfer signal.
13. The safety injection pumps prematurely transferred their suction source from to Boric Acid Storage Tank to the RWST.


Reactor at Full Power
Loss of all 115kv transmission lines due to maintenance, a ground fault, and power swings
Loss of auxiliary power source due to overloading
Reactor trip due to false overpower/ΔT signal
Emergency diesel generators start
AFW pumps start
Safety injection due to excessive cooldown
Momentary loss of power to vital instrument bus 1A
Boric Acid Storage Tank level indicators powered by bus 1A fail downscale
Vital instrument bus 1A power restored
"Lo-lo" signal generated by failed-low Boric Acid Storage Tank level indicators before indicators returned to normal values, resulting in premature safety injection pump transfer signal and subsequent safety injection pump premature tranfer from Boric Acid Storage Tank to RWST.
Potential Severe Core Damage
No - concentrated boric acid not required
No
No - redundant instrument power available
No
No
No - excessive long term cooldown prevented
(all AFW pumps fail)
Yes
No
(turbine driven AFW pump)
Yes
(all AFW pumps fail)
ATWS
No
No

NSIC 85370 - Actual Occurrence for Loss of Offsite Power, Excessive RCS Cooldown, Safety Injection, and Failure of a Vital Instrument Bus at R. E. Ginna


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
(*)
Yes 9
No 10
Yes 11
No 12
Yes 13

NSIC 85370 — **Sequence of Interest for Loss of Offsite Power at R. E. Ginna**

*Use of HPI following AFW failure not included in mitigation procedures.

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 85370

DATE OF LER: October 31, 1973

DATE OF EVENT: October 21, 1973

SYSTEM INVOLVED: offsite power, vital power

COMPONENT INVOLVED: transmission circuits, vital instrument bus

CAUSE: (see attached page)

SEQUENCE OF INTEREST: loss of offsite power

ACTUAL OCCURRENCE: loss of offsite power, excessive RCS cooldown, subsequent safety injection, and loss of a vital instrument bus

REACTOR NAME: R. E. Ginna

DOCKET NUMBER: 50-244

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 490 MWe

REACTOR AGE: 4.1 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Gilbert Associates

OPERATORS: Rochester, Gas & Electric Co.

LOCATION: 15 miles NE of Rochester, NY

DURATION: N/A

PLANT OPERATING CONDITION: approximately 100% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; [✓ (d)] premature performance

DISCOVERY METHOD: operational event

COMMENT: –

[redacted]

CAUSE: The transmission circuits failed due to a ground fault and power swings caused by instability of the generator and transmission lines when two transmission lines were inoperable. The cause of the momentary power failure on the vital instrument bus was not determined.

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 85566

Date:

Title: Complete Loss of a.c. Power Causes RCIC and HPCI to be Inoperable.

The failure sequence was:

1. In preparation for the turbogenerator trip and loss of offsite power test, the 4 kv unit boards were placed in maual to prevent automatic transfer.
2. The turbine was manually tripped because of increased vibration. This resulted in a scram since offsite power could no longer be supplied, (because of 1.)
3. The RCIC and the HPCI systems could not be started until the standby diesels were energized, since their rest logic required a.c. power.

Corrective action:

The power source for some RCIC/HPCI relays was changed and the manual reset was eliminated.

Design purpose of failed system or component:

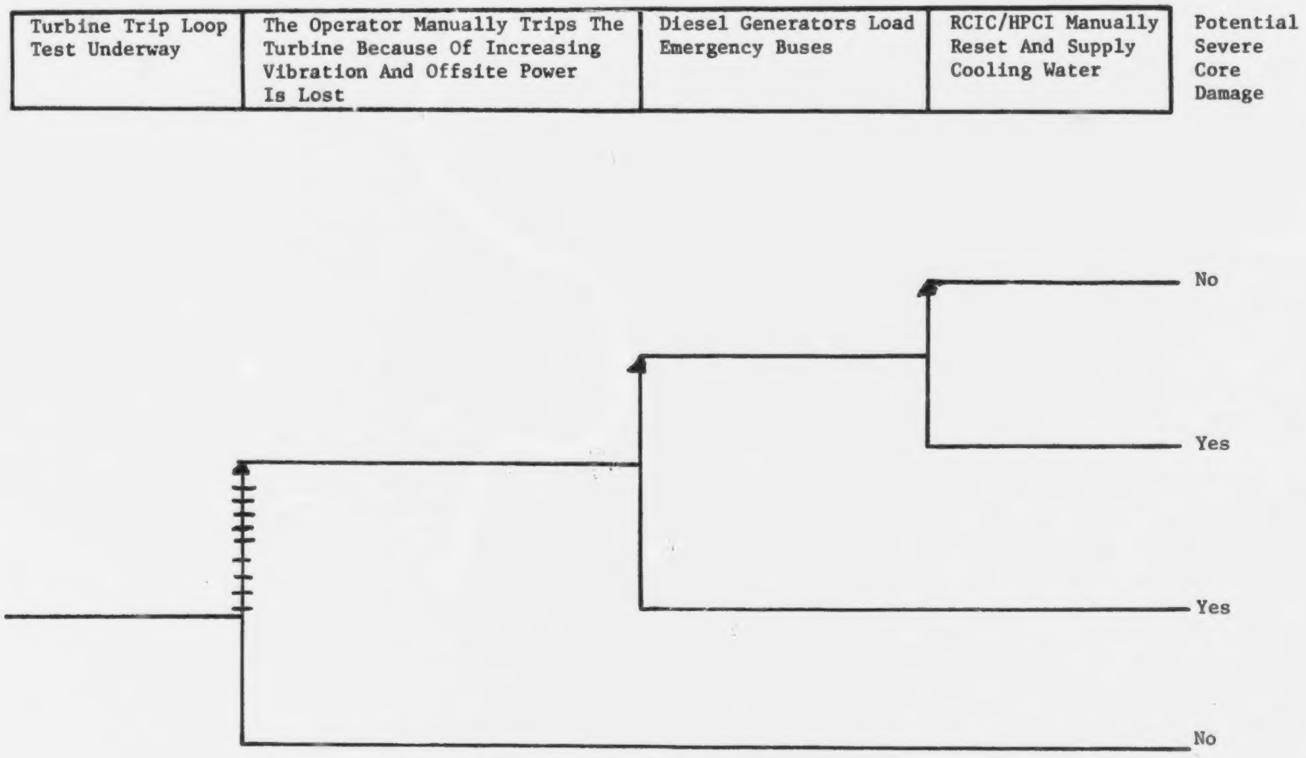
1. RCIC provides a source of water should the reactor become isolated.
2. HPCI provides a high pressure coolant supply during a small LOCA.

Unavailability of system per WASH 1400:* RCIC: 8.0×10^{-2}/D
HDCI: 8.8×10^{-2}/D

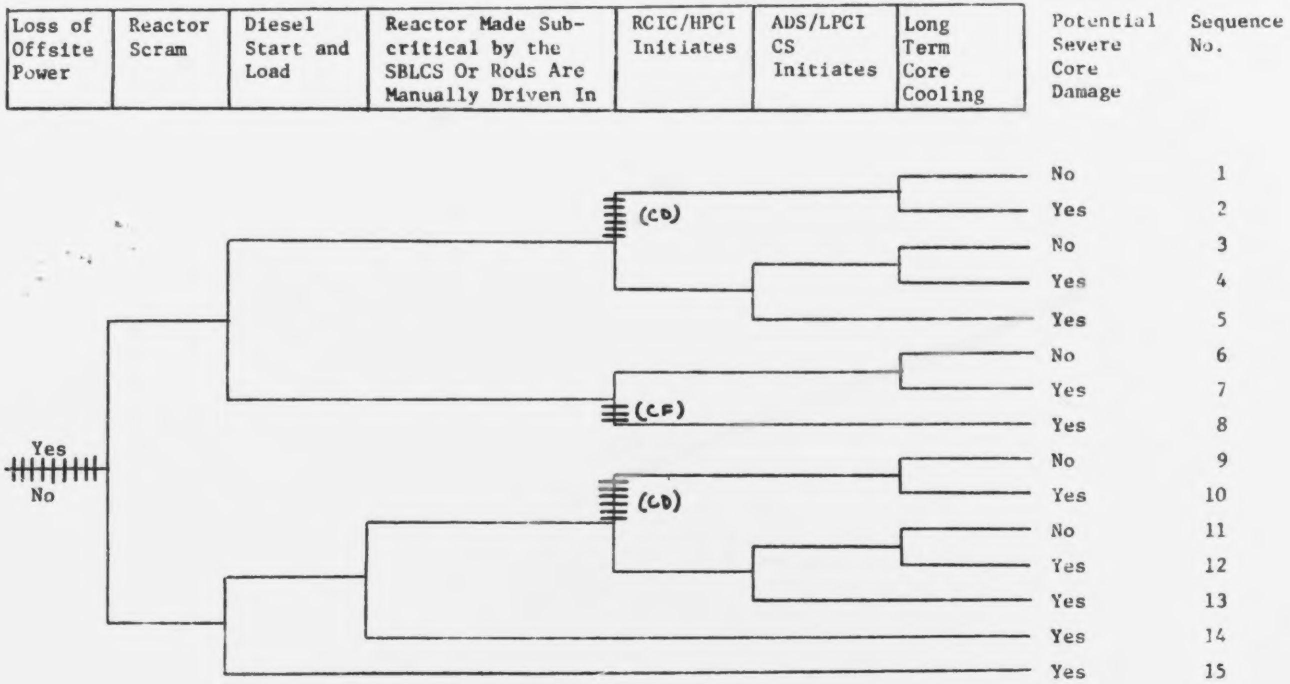
Unavailability of component per WASH 1400:*

---

[*] Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Turbine Trip Loop Test Underway
The Operator Manually Trips The Turbine Because Of Increasing Vibration And Offsite Power Is Lost
Diesel Generators Load Emergency Buses
RCIC/HPCI Manually Reset And Supply Cooling Water
Potential Severe Core Damage
No
Yes
Yes
No

NSIC 85566 — Actual Occurrence of Complete Loss of Plant A.C. Power Causes RCIC and HPCI to be Unavailable


Loss of Offsite Power
Reactor Scram
Diesel Start and Load
Reactor Made Subcritical by the SBLCS Or Rods Are Manually Driven In
RCIC/HPCI Initiates
ADS/LPCI CS Initiates
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
(CD)
(CF)
(CD)
No 1
Yes 2
No 3
Yes 4
Yes 5
No 6
Yes 7
Yes 8
No 9
Yes 10
No 11
Yes 12
Yes 13
Yes 14
Yes 15

NSIC 85566 — Sequence of Interest for Complete Loss of Plant A.C. Power Caused RCIC and HPCI to be Inoperable at Browns Ferry 1

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 85566

DATE OF LER: November 19, 1973

DATE OF EVENT: –

SYSTEM INVOLVED: RCIC/HPCI

COMPONENT INVOLVED: RCIC/HPCI tripping logic

CAUSE: Design error

SEQUENCE OF INTEREST: Loss of Feedwater Flow

ACTUAL OCCURRENCE: Complete Loss of a.c. Power Causes RCIC and HPCI to be inoperable.

REACTOR NAME: Browns Ferry 1

DOCKET NUMBER: 259

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 1065 MWe

REACTOR AGE: .18 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: TVA

OPERATORS: TVA

LOCATION: 10 miles NW of Decatur, AL

DURATION: N/A

PLANT OPERATING CONDITION: Just scrammed during testing

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ________________

DISCOVERY METHOD: Operational Event.

COMMENT: Common Mode Design Error, a.c. requirement.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 85738

Date:

Title: RCIC and HPCI Failure During Testing at Browns Ferry 1

The failure sequence was:

1. During startup testing the RCIC system failed to operate when manually initiated. The steam supply valve failed to open.
2. HPCI was then manually initiated to maintain vessel water level. However, it isolated and tripped.
3. The operator reset the isolation circuit and reinitiated HPCI which operated successfully.

Corrective action:

1. The RCIC steam supply valve was repaired.
2. No cause for the HPCI trip could be found.

Design purpose of failed system or component:

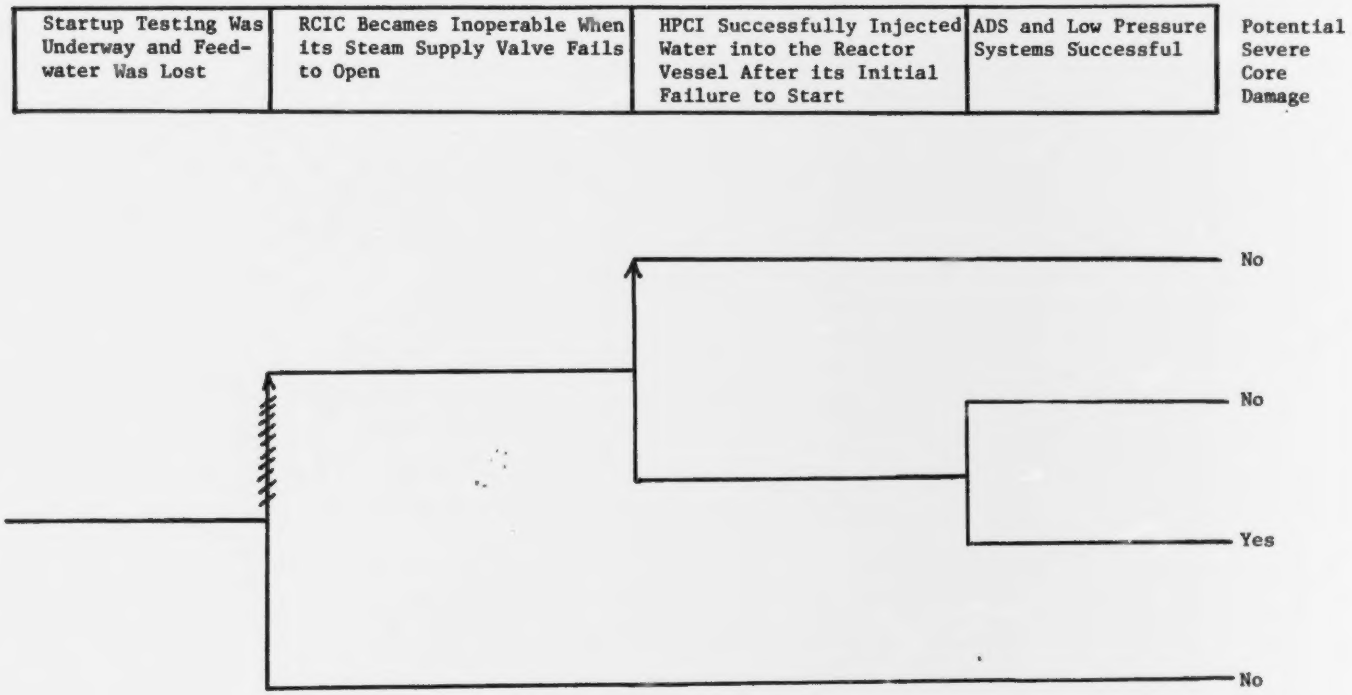
1. RCIC provides a source of water should the reactor become isolated.
2. HPCI provides a high pressure coolant supply during a small LOCA.

Unavailability of system per WASH 1400:* RCIC 8.0×10^{-2}/D
HPIC 8.8×10^{-2}/D

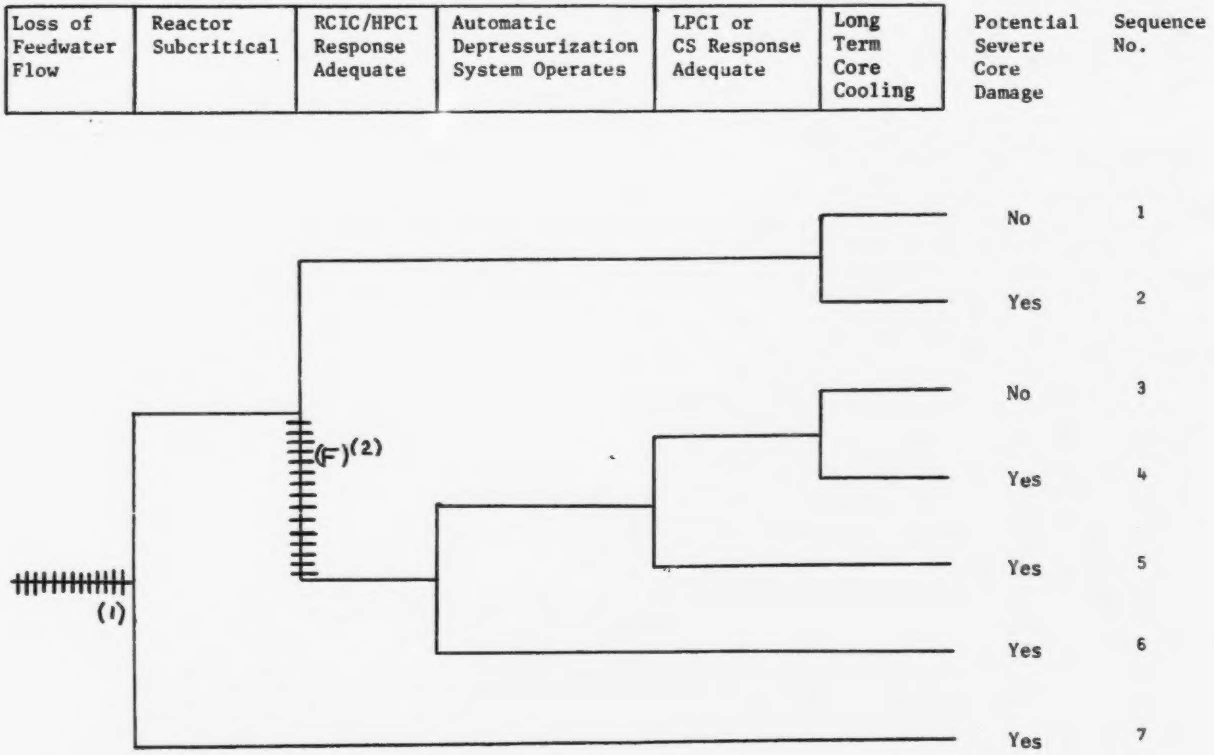
Unavailability of component per WASH 1400:*

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Startup Testing Was Underway and Feed-water Was Lost
RCIC Becames Inoperable When its Steam Supply Valve Fails to Open
HPCI Successfully Injected Water into the Reactor Vessel After its Initial Failure to Start
ADS and Low Pressure Systems Successful
Potential Severe Core Damage
No
No
Yes
No

NSIC 85738 — Actual Occurrence of RCIC and HPCI Failure During Testing at Browns Ferry 1


Loss of Feedwater Flow
Reactor Subcritical
RCIC/HPCI Response Adequate
Automatic Depressurization System Operates
LPCI or CS Response Adequate
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
(1)
(F)(2)
No
1
Yes
2
No
3
Yes
4
Yes
5
Yes
6
Yes
7

NSIC 85738 — Sequence of Interest for RCIC and HPCI Failures During Testing at Browns Ferry 1

[1] An apparent loss of feedwater.

[2] HPCI was manually reset and operated satisfactorily.

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 85738

DATE OF LER: November 19, 1973

DATE OF EVENT:

SYSTEM INVOLVED: RCIC/HPCI

COMPONENT INVOLVED: Steam supply valve, Isolation circuits

CAUSE: Mechanical Failure

SEQUENCE OF INTEREST: Loss of Feedwater

ACTUAL OCCURRENCE: HPCI and RCIC Failures during testing.

REACTOR NAME: Browns Ferry 1

DOCKET NUMBER: 259

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 1065 MWe

REACTOR AGE: .18 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: TVA

OPERATORS: TVA

LOCATION: 10 miles NW of Decatur, AL

DURATION: N/A

PLANT OPERATING CONDITION: Start up testing

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; [✓ (b)] failed to start; (c) made inoperable; (d) ____________

DISCOVERY METHOD: Operational Event while startup testing.

COMMENT: No LER

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 86990

Date: November 17, 1973

Title: Loss of Incoming Power at Nine Mile Point from a Worker Bumpkng a Relay

The failure sequence was:

1. The reactor was shutdown for an AEC operator's examination involving a critical demonstration.
2. After the reactor was shutdown, breaker R40 was opened to facilitate line work on the 115 Kv line, breaker R10, the remaining 115 kV line, was closed and the station was operating on reserve power supplied by this line.
3. A plant electrician working in the auxiliary control room bumped a relay which caused relay 9 and 5 to de-energize thus trapping the remaining 115 kV line.
4. The plant remained without offsite power for 10 seconds until break R10 was reclosed.

Corrective action:

The R10 breaker was reclosed within 10 seconds.

Design purpose of failed system or component:

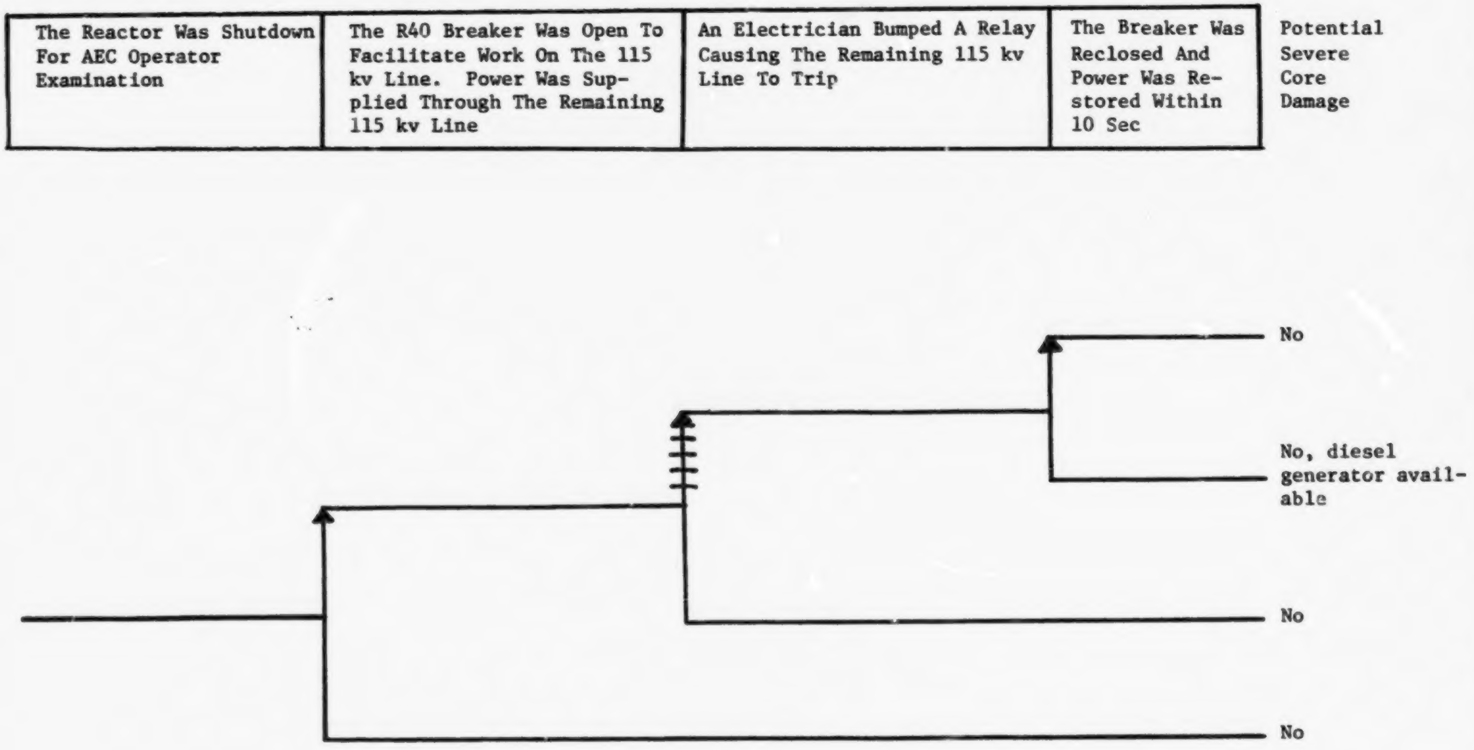
Offsite power supplies house loads when the generator is down.

Unavailability of system per WASH 1400:* LOOP: 2×10^{-5}/Hr

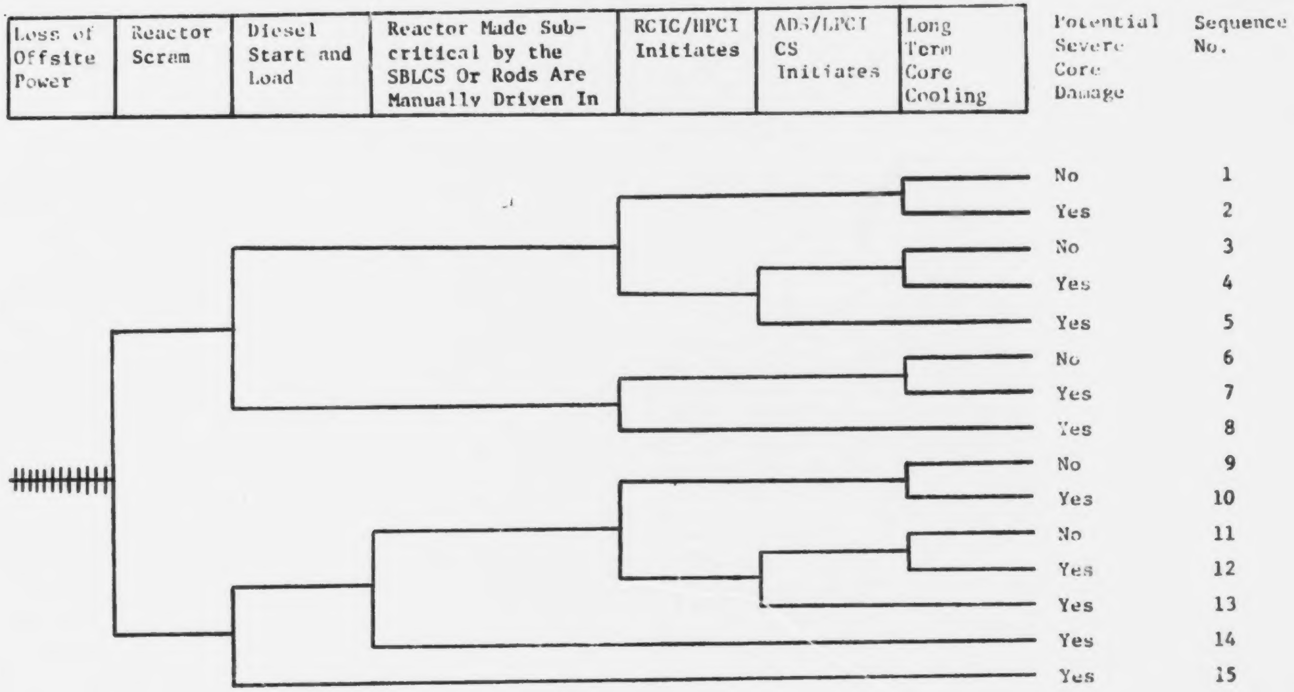
Unavailability of component per WASH 1400:* -

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


The Reactor Was Shutdown For AEC Operator Examination
The R40 Breaker Was Open To Facilitate Work On The 115 kv Line. Power Was Supplied Through The Remaining 115 kv Line
An Electrician Bumped A Relay Causing The Remaining 115 kv Line To Trip
The Breaker Was Reclosed And Power Was Restored Within 10 Sec
Potential Severe Core Damage
No
No, diesel generator available
No
No

NSIC 86990 — Actual Occurrence of Loss of Incoming Power at Nine Mile Point from a Worker Bumping a Relay


Loss of Offsite Power
Reactor Scram
Diesel Start and Load
Reactor Made Subcritical by the SBLCS Or Rods Are Manually Driven In
RCIC/HPCI Initiates
ADS/LPCI CS Initiates
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
Yes 2
No 3
Yes 4
Yes 5
No 6
Yes 7
Yes 8
No 9
Yes 10
No 11
Yes 12
Yes 13
Yes 14
Yes 15

NSIC 86990 — Sequence of Interest of Loss of Incoming Power of Nine Mile Point from Worker Bumping Relay

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 86990

DATE OF LER: November 21, 1973

DATE OF EVENT: November 11, 1973

SYSTEM INVOLVED: Offsite power

COMPONENT INVOLVED: Breakers

CAUSE: Operator

SEQUENCE OF INTEREST: Loss of Offsite Power

ACTUAL OCCURRENCE: Loss of Incoming

REACTOR NAME: Nine Mile Point

DOCKET NUMBER: 220

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 72 MWe

REACTOR AGE: 5.2 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Consumer Power Corporation

LOCATION: Eight miles NE of Oswego, NY

DURATION: N/A

PLANT OPERATING CONDITION: just critical

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ______________________

DISCOVERY METHOD: Operational transient

COMMENT: A station blackout loop

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 88451

Date: February 1, 1974

Title: Total Loss of Station Service Power and Failure of Service Water Pumps for One Diesel-Generator to Start at Connecticut Yankee

The failure sequence was:

1. With the reactor at power during an ice storm, one of the two transmission lines providing station service power (line 772) tripped due to a faulted lightning arrestor on an adjacent line.
2. The second of two transmission lines providing station service power (line 1206) tripped due to improper blocking relay operation.
3. Both diesel generators started.
4. Diesel-generator A service water pump failed to start due to too rapid actuation of the pump breaker undervoltage devices.
5. Diesel-generator A service water pump was manually started.

Corrective action:

1. The exact cause of the improper blocking relay operation was still under investigation at the time the LER was written.
2. The time delay on the undervoltage device associated with the diesel-generator service water pumps was reset.

Design purpose of failed system or component:

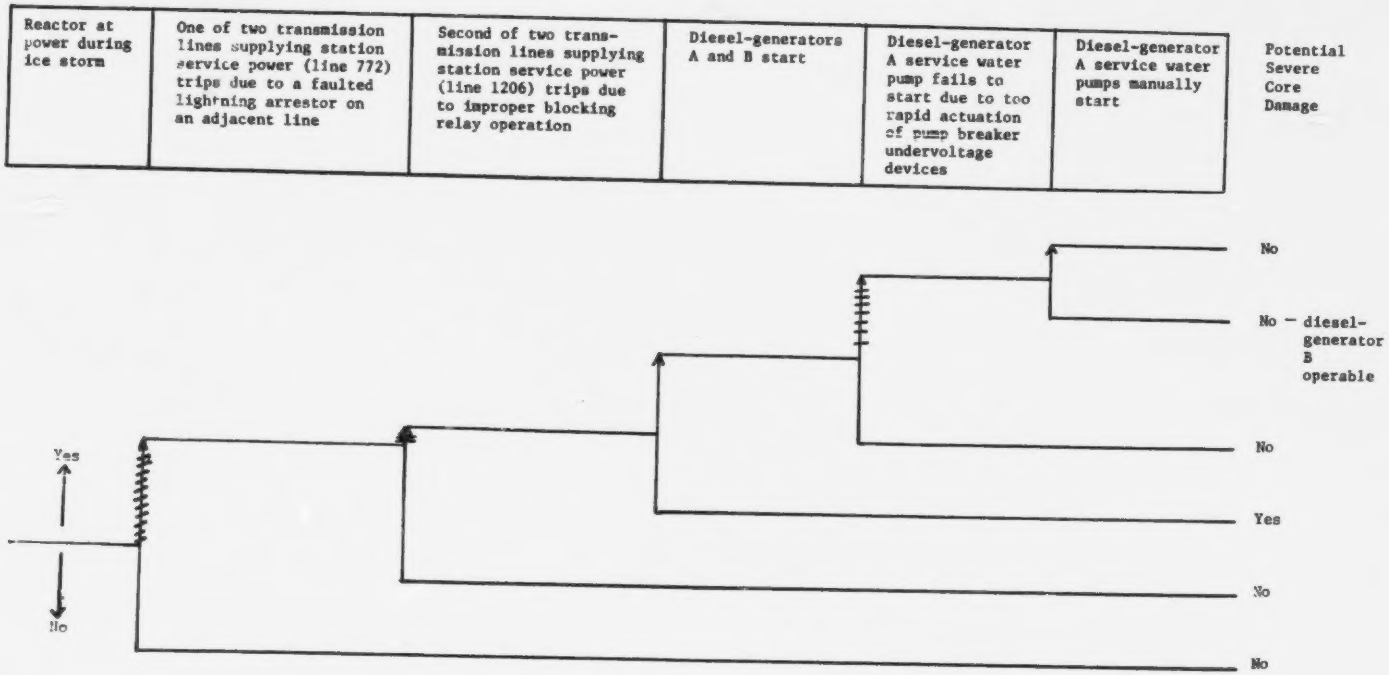
1. Station service power provides normal power to the station auxiliary loads.
2. The diesel-generator service water pumps provide cooling water for the diesel-generator during operation.

Unavailability of system per WASH 1400:* offsite power: 2×10^{-5}/hr

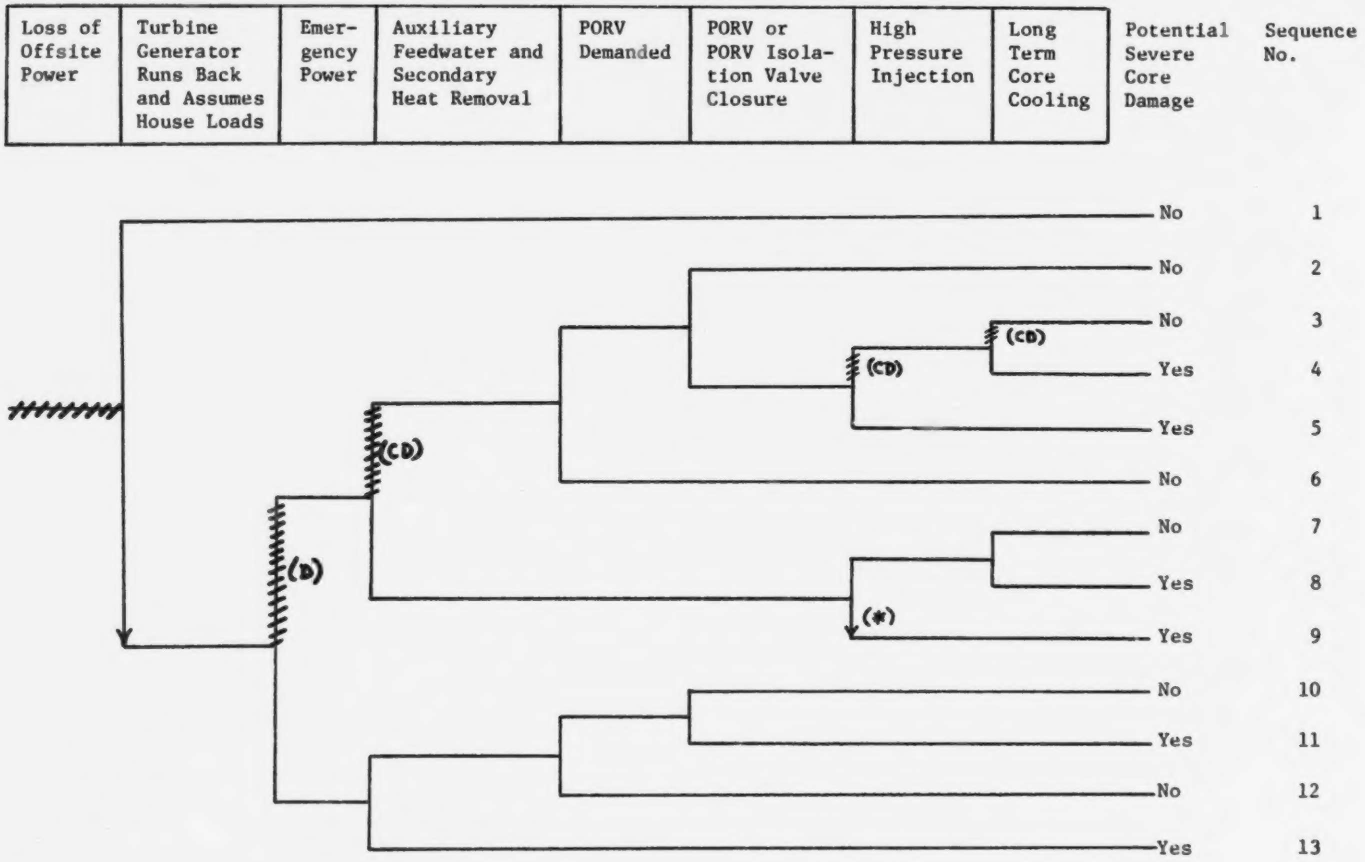
Unavailability of component per WASH 1400:* Diesel-generator: 3×10^{-2}/D

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor at power during ice storm
One of two transmission lines supplying station service power (line 772) trips due to a faulted lightning arrestor on an adjacent line
Second of two transmission lines supplying station service power (line 1206) trips due to improper blocking relay operation
Diesel-generators A and B start
Diesel-generator A service water pump fails to start due to too rapid actuation of pump breaker undervoltage devices
Diesel-generator A service water pumps manually start
Potential Severe Core Damage
Yes
No
No
No - diesel-generator B operable
No
Yes
No
No

NSIC 88451 — Actual Occurrence for Total Loss of Station Service Power and Failure to Start of Service Water Pump for One Diesel-Generator at Connecticut Yankee


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
(D)
(CD)
(CD)
(CD)
(*)
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
Yes 9
No 10
Yes 11
No 12
Yes 13

NSIC 88451 — Sequence of Interest for Total Loss of Station Service Power and Failure of Service Water Pump for One Diesel-Generator at Connecticut Yankee

*Use of HPI following loss of AFW was not included in mitigation procedures.

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 88451

DATE OF LER: February 1, 1974

DATE OF EVENT: January 19, 1974

SYSTEM INVOLVED: Station service power (offsite power) emergency power

COMPONENT INVOLVED: Transmission lines

CAUSE: Both transmission lines tripped due to a faulted lightning arrestor and an improperly operating blocking relay; the service water pump failed to start due to a too rapidly operating under-voltage device.

SEQUENCE OF INTEREST: Loss of Offsite Power

ACTUAL OCCURRENCE: Loss of Offsite Power

REACTOR NAME: Haddam Neck

DOCKET NUMBER: 50-213

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 575 MWe

REACTOR AGE: 6.7 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Stone and Webster

OPERATORS: Connecticut Yankee Atomic Power Co.

LOCATION: 13 miles east of Meridan, Conn.

DURATION: N/A

PLANT OPERATING CONDITION: at power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c) made inoperable]; (d) ____________

DISCOVERY METHOD: during operation

COMMENT: —

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 89153

Date: March 1, 1974

Title: Inadvertant Opening of RWST Isolation Valves at Prairie Island 1

The failure sequence was:

1. With the reactor at 50% power, the isolation valves to the Safety Injection Pumps from the RWST were found in the open position due to an inconsistency between the previously conducted boric acid tank level indication system monthly test procedure and the RWST isolation valve "open" logic. Such a valve position would have prevented injection of concentrated boric acid following a steam line break.

Corrective action:

The valves were closed and the test procedure revised to prevent opening the valves.

Design purpose of failed system or component:

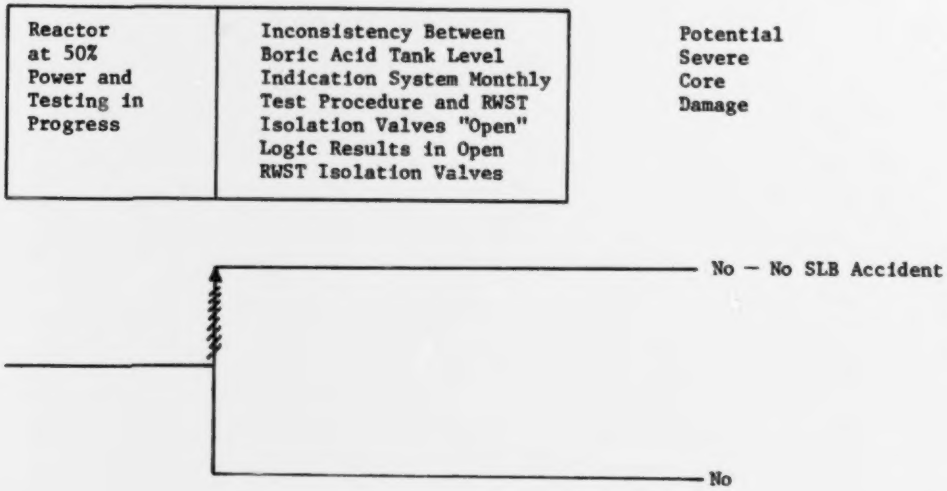
The normally closed valves permit the injection of concentrated boric acid into the RCS following a steam line break prior to the injection of normal RWST water. Such injection terminates the return to power which occurs following a steam line break when the most reactive control rod fails to insert.

Unavailability of system per WASH 1400:* SLB accident not specifically considered in WASH-1400.

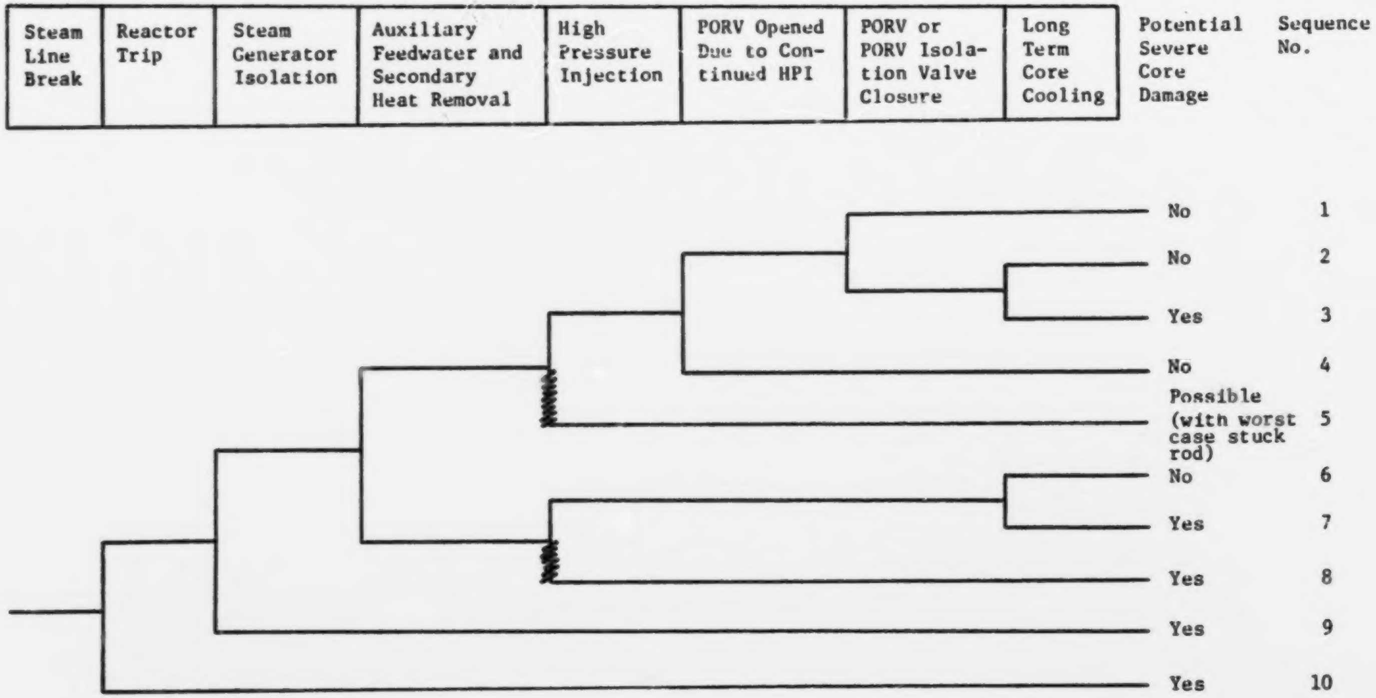
Unavailability of component per WASH 1400:*

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor at 50% Power and Testing in Progress
Inconsistency Between Boric Acid Tank Level Indication System Monthly Test Procedure and RWST Isolation Valves "Open" Logic Results in Open RWST Isolation Valves
Potential Severe Core Damage
No — No SLB Accident
No

NSIC 89153 — Actual Occurrence for Inadvertant Opening of RWST Isolation Valves at Prairie Island 1


Steam Line Break
Reactor Trip
Steam Generator Isolation
Auxiliary Feedwater and Secondary Heat Removal
High Pressure Injection
PORV Opened Due to Continued HPI
PORV or PORV Isolation Valve Closure
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
Yes 3
No 4
Possible (with worst case stuck rod) 5
No 6
Yes 7
Yes 8
Yes 9
Yes 10

NSIC 89153 — Sequence of Interest for Inadvertant Opening of RWST Isolation Valves at Prairie Island 1

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 89153

DATE OF LER: March 1, 1974

DATE OF EVENT: February 21, 1974

SYSTEM INVOLVED: Safety Injection

COMPONENT INVOLVED: RWST Isolation Valves

CAUSE: Valves inadvertantly opened during testing

SEQUENCE OF INTEREST: Steam line break

ACTUAL OCCURRENCE: Open RWST isolation valves

REACTOR NAME: Prairie Island 1

DOCKET NUMBER: 50-282

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 530 MWe

REACTOR AGE: .22 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Pioneer Services

OPERATORS: Northern States Power Company

LOCATION: 28 miles SE of Minneapolis, Minnesota

DURATION: 24(a) hours
PLANT OPERATING CONDITION: 50% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________

DISCOVERY METHOD: Operator observation

COMMENT:

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: **89205**

Date: **November 30, 1973**

Title: **Failure of Reactor Coolant Pump Shaft at Surry 1**

The failure sequence was:

1. With the reactor at 95% power, operations personnel noted reactor coolant flow fluctuations varying between 96-100% and a 12 mil (peak-peak) vibration of reactor coolant pump (RCP) A.
2. A controlled shutdown was initiated.
3. At 42% power the reactor tripped coincident with a sudden loss of flow in loop A, which resulted from the failure of the RCP A shaft between the thermal barrier and the lower radial bearing.

Corrective action:

1. The pump was disassembled and the shaft removed for examination. Westinghouse and Battelle concluded the failure was caused by fatigue at an improperly machined groove on the shaft.
2. A more recently manufactured shaft was installed in the pump.

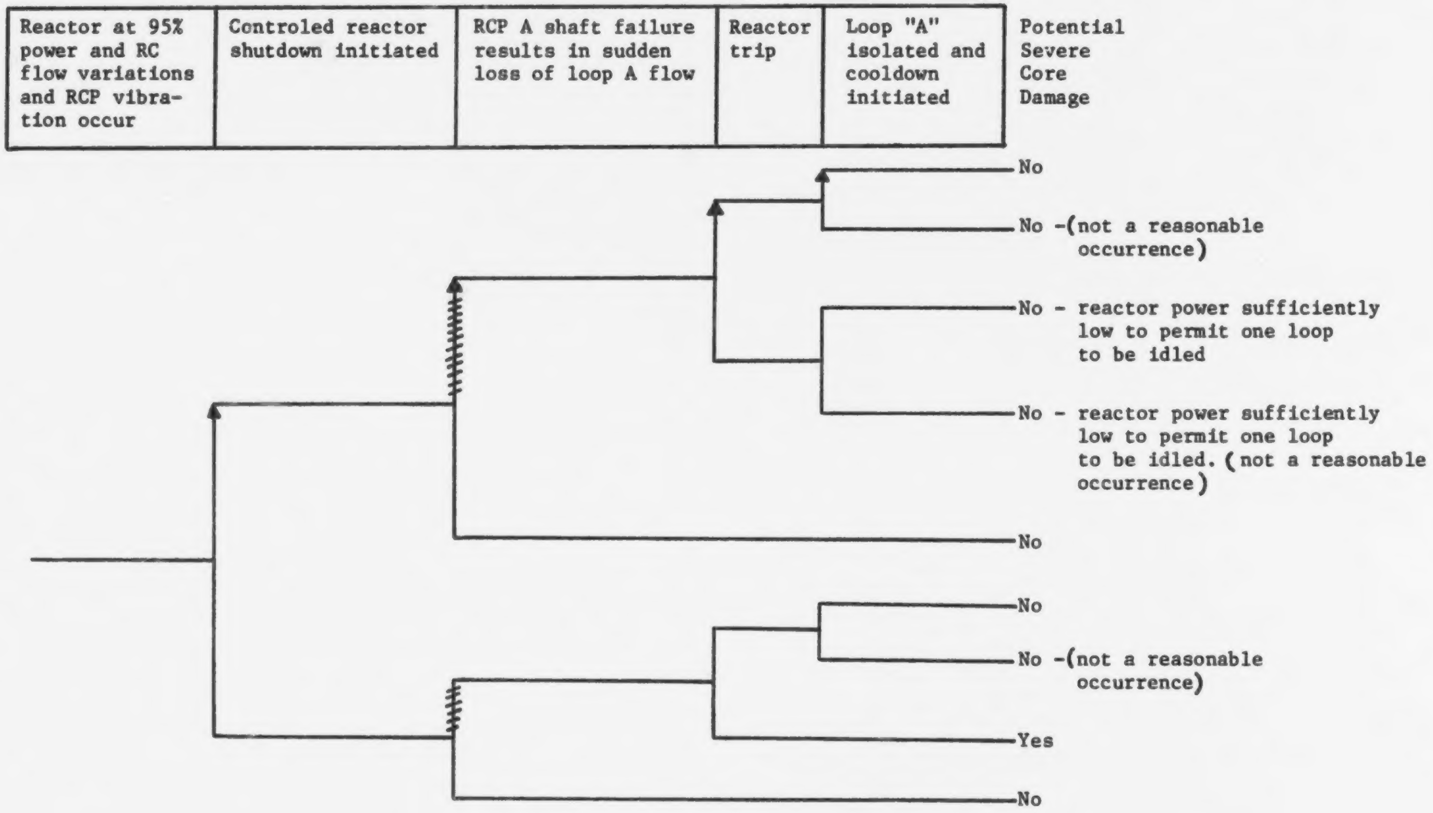
Design purpose of failed system or component:
The reactor coolant pumps provide forced reactor coolant flow through the core and steam generators.

Unavailability of system per WASH 1400:* **N.A.**

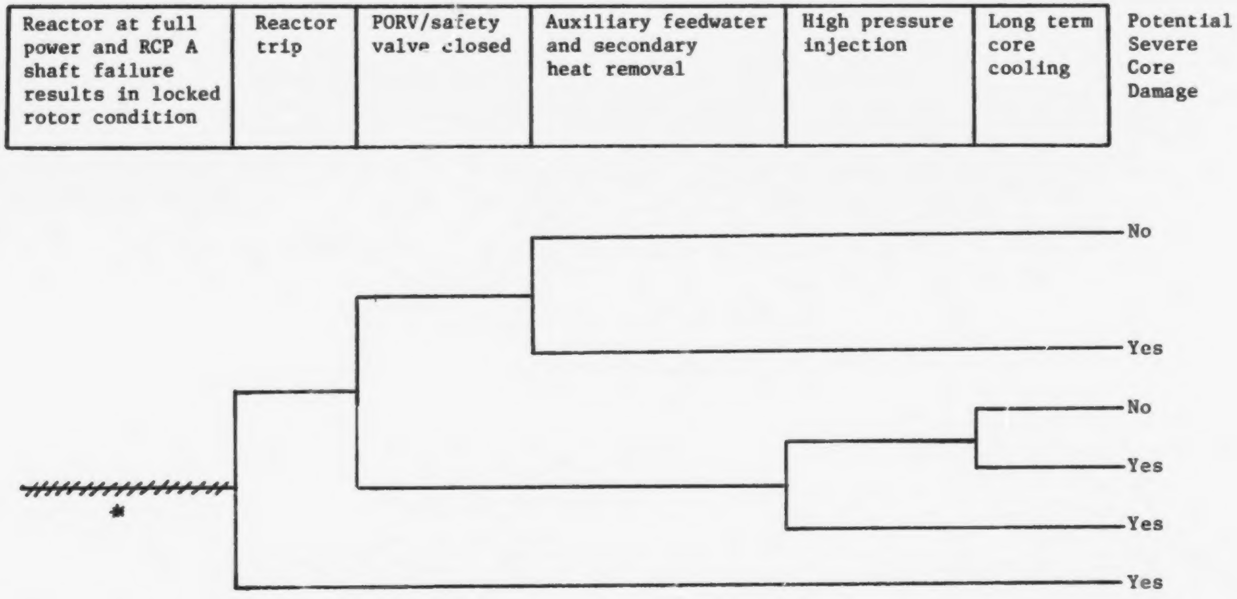
Unavailability of component per WASH 1400:* **pump, failure to run: 3×10^{-4}/hr.**

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor at 95% power and RC flow variations and RCP vibration occur
Controled reactor shutdown initiated
RCP A shaft failure results in sudden loss of loop A flow
Reactor trip
Loop "A" isolated and cooldown initiated
Potential Severe Core Damage
No
No -(not a reasonable occurrence)
No - reactor power sufficiently low to permit one loop to be idled
No - reactor power sufficiently low to permit one loop to be idled. (not a reasonable occurrence)
No
No
No -(not a reasonable occurrence)
Yes
No

NSIC 89205 — Actual Occurrence for Failure of a Reactor Coolant Pump Shaft at Surry 1


Reactor at full power and RCP A shaft failure results in locked rotor condition
Reactor trip
PORV/safety valve closed
Auxiliary feedwater and secondary heat removal
High pressure injection
Long term core cooling
Potential Severe Core Damage
*
No
Yes
No
Yes
Yes
Yes

NSIC 89205 — Sequence of Interest for Failure of a Reactor Coolant Pump Shaft at Surry 1

*Surry pump failure at full power instead of at 42% power and with perhaps a more abrupt stopping of the pump impeller.

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 89205

DATE OF LER: February 24, 1974

DATE OF EVENT: November 30, 1973

SYSTEM INVOLVED: Reactor Coolant System

COMPONENT INVOLVED: Reactor Coolant Pump

CAUSE: Failed shaft

SEQUENCE OF INTEREST: Locked RCP Rotor at Full Power

ACTUAL OCCURRENCE: Failed RCP Shaft at 42% Power

REACTOR NAME: Surry 1

DOCKET NUMBER: 50-280

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 822 MWe

REACTOR AGE: 1.4 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Stone & Webster

OPERATORS: Virginia Electric and Power Co.

LOCATION: 17 mi Northwest of Newport News, Va.

DURATION: N/A

PLANT OPERATING CONDITION: 42% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c) made inoperable]; (d) ______________________

DISCOVERY METHOD: operator observation

COMMENT: A reactor coolant isolation valve shaft also failed during the shutdown for this incident.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: **89252**

Date: **February 17, 1974**

Title: **Three Related Abnormal Occurrences with Diesel Generator at Zion 2**

The failure sequence was:

1. **Two auxiliary feed pumps were out of service.**
2. **During testing of the diesel generator associated with the operable auxiliary feed pump, the diesel generator tripped on overspeed while starting.**
3. **Once the diesel was started, its output breaker would not close on the associated safety-related bus because of a relay failure.**

Corrective action:

1. **The synchro-speed setter was readjusted to its correct setting, which permitted the diesel to start without tripping out on overspeed.**
2. **A foreign object (screw and two washers) was removed from the failed relay and the relay was returned to service.**

Design purpose of failed system or component:

1. **The diesel generators provide emergency electric power to safety related loads in the event of loss of offsite power.**
2. **The failed relay permits generator output breaker closure once the generator is up to frequency.**

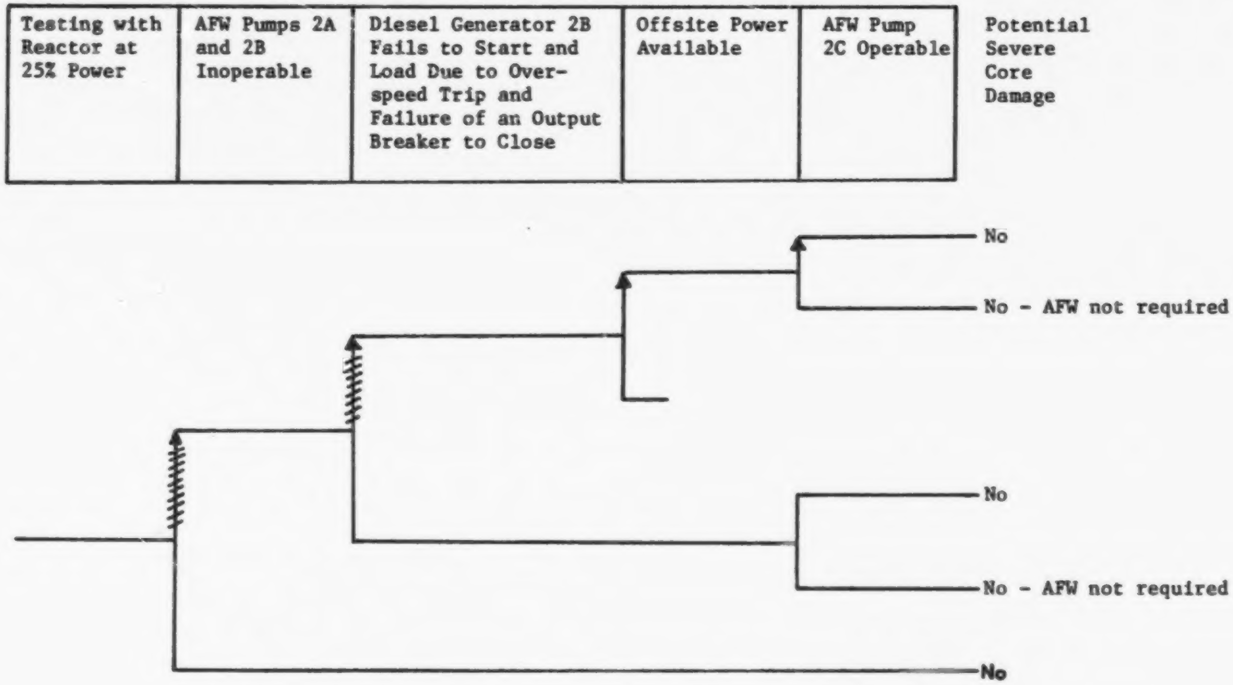
Unavailability of system per WASH 1400:* **Emergency Electric Power: 1×10^{-2}/D**

Unavailability of component per WASH 1400:* **Diesel Generator: 3×10^{-2}/D**
Relay, failure to energize: 1×10^{-4}/D

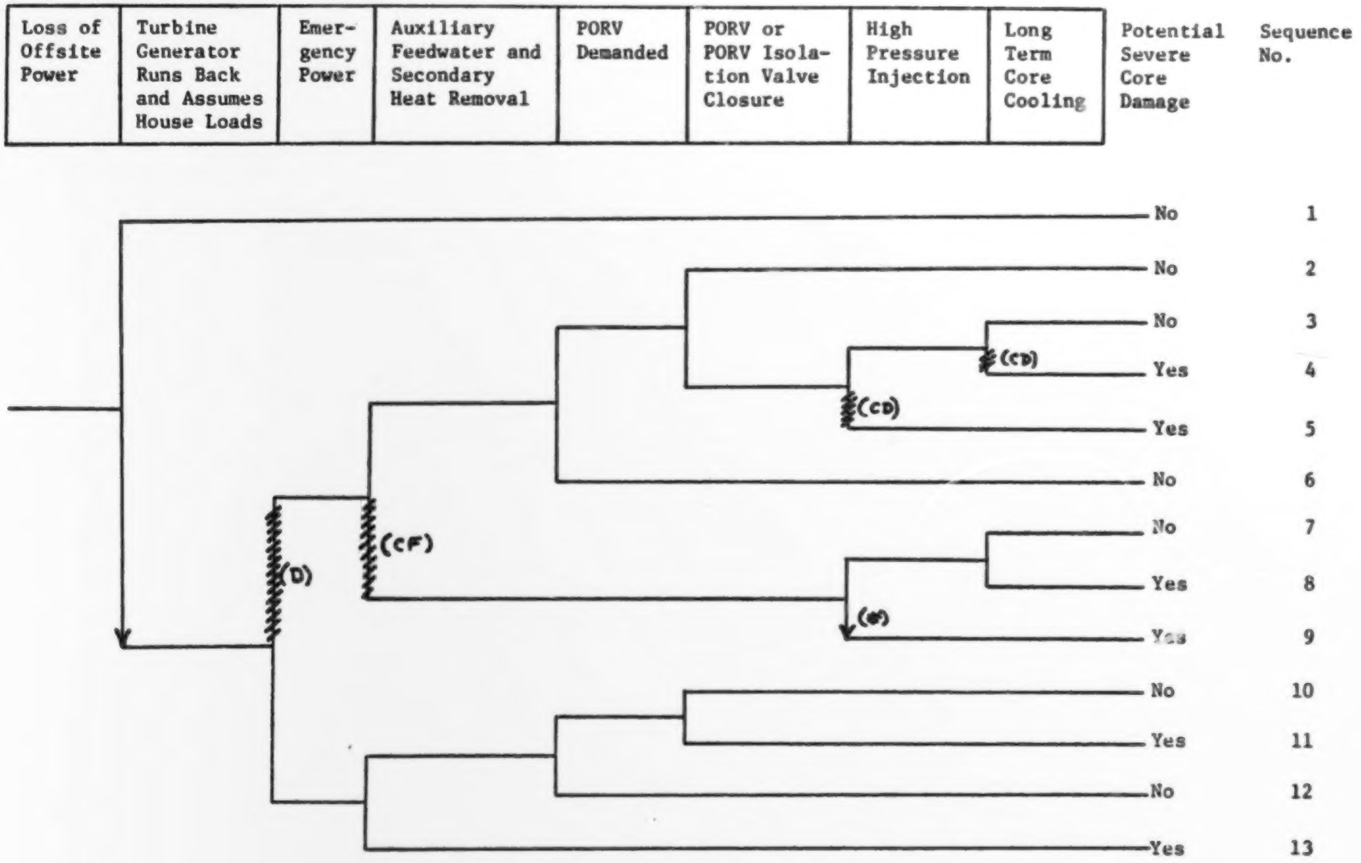
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

BEST DOCUMENT AVAILABLE


Testing with Reactor at 25% Power
AFW Pumps 2A and 2B Inoperable
Diesel Generator 2B Fails to Start and Load Due to Overspeed Trip and Failure of an Output Breaker to Close
Offsite Power Available
AFW Pump 2C Operable
Potential Severe Core Damage
No
No - AFW not required
No
No - AFW not required
No

NSIC 89252 — Actual Occurrence for Failure of Diesel Generator to Start and Load When Two AFW Pumps Were Out of Service at Zion 2


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
(D)
(CF)
(CD)
(CD)
(*)
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
Yes 9
No 10
Yes 11
No 12
Yes 13

NSIC 89252 — Sequence of Interest for Failure of Diesel Generator to Start and Load When Two AFW Pumps Were Out of Service at Zion 2

*Use of HPI following AFW failure not included in mitigating procedures.

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 89252

DATE OF LER: February 27, 1974

DATE OF EVENT: February 17, 1974

SYSTEM INVOLVED: Emergency electric power and auxiliary feedwater

COMPONENT INVOLVED: Diesel generator, diesel generator output breaker, and AFW pumps

CAUSE: The diesel generator failed to start due to incorrect synchro-speed setting, an output breaker relay failed to close because of a foreign object lodged in the relay, and two AFW pumps were inoperable (reason not stated).

SEQUENCE OF INTEREST: loss of offsite power

ACTUAL OCCURRENCE: Failure of diesel-generator to start and load when 2 AFW pumps were out-of-service.

REACTOR NAME: Zion 2

DOCKET NUMBER: 50-304

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 1040 MWe

REACTOR AGE: .15 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Sargent & Lundy

OPERATORS: Commonwealth Edison Co.

LOCATION: 40 miles north of Chicago, Ill.

DURATION: 6.75 hours

PLANT OPERATING CONDITION: 25% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; [✓ (b)] failed to start; [✓ (c)] made inoperable; (d) ____________________

DISCOVERY METHOD: during testing

COMMENT: –

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 90421

Date: April 17, 1974

Title: Auxiliary Feed Pump Suction Strainers Clogged at Point Beach 1

The failure sequence was:

1. With Unit 1 in a cooldown, the "A" motor-driven AFW pump did not provide adequate flow.
2. An in-line suction strainer, of which the operators were unaware, was found to be 95% plugged.
3. A similar strainer, similarly plugged, was found in the suction line of the "B" motor-driven AFW pump.
4. A partially plugged strainer was found in the suction lines of both units turbine-driven pumps.

Corrective action:

1. All the strainers were removed.
2. Other piping systems were examined to determine if other unidentified strainers existed.

Design purpose of failed system or component:

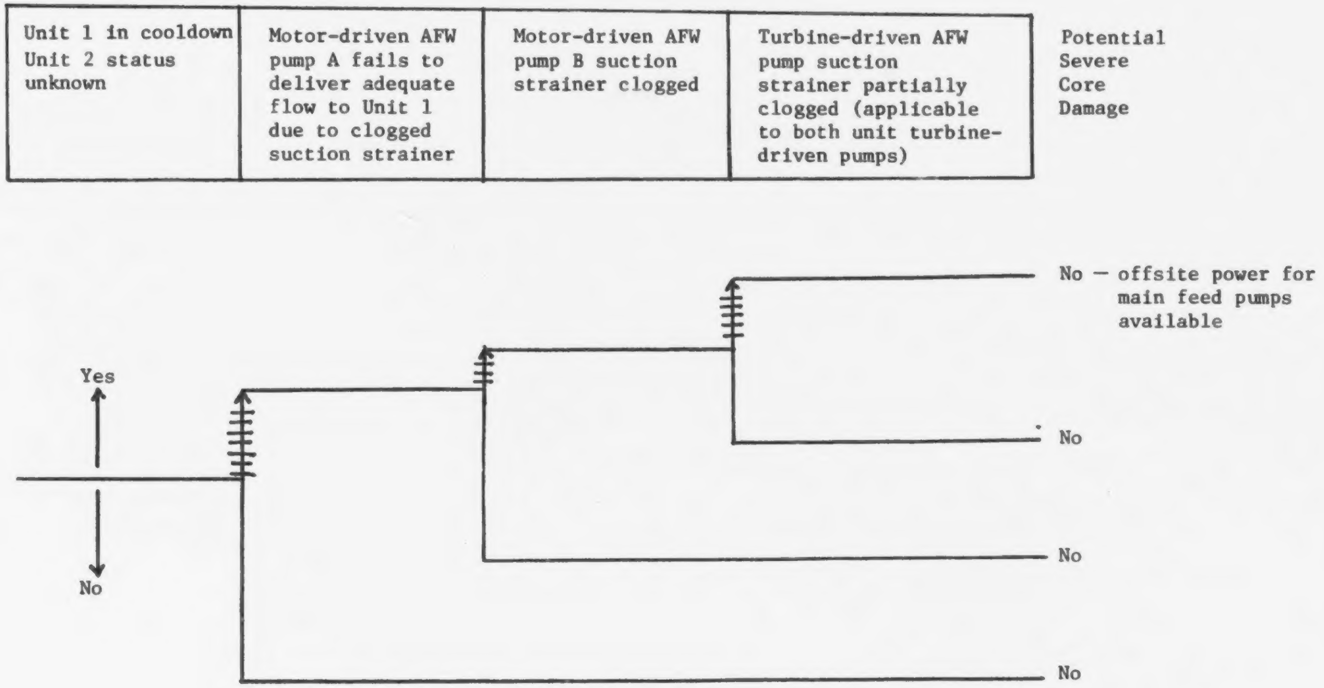
1. The auxiliary feedwater pumps provide water to the steam generators for reactor cooling in the event the main feedwater system is not operable.

Unavailability of system per WASH 1400:* auxiliary feedwater: 2.5×10^{-4}
(start + 8 hours)

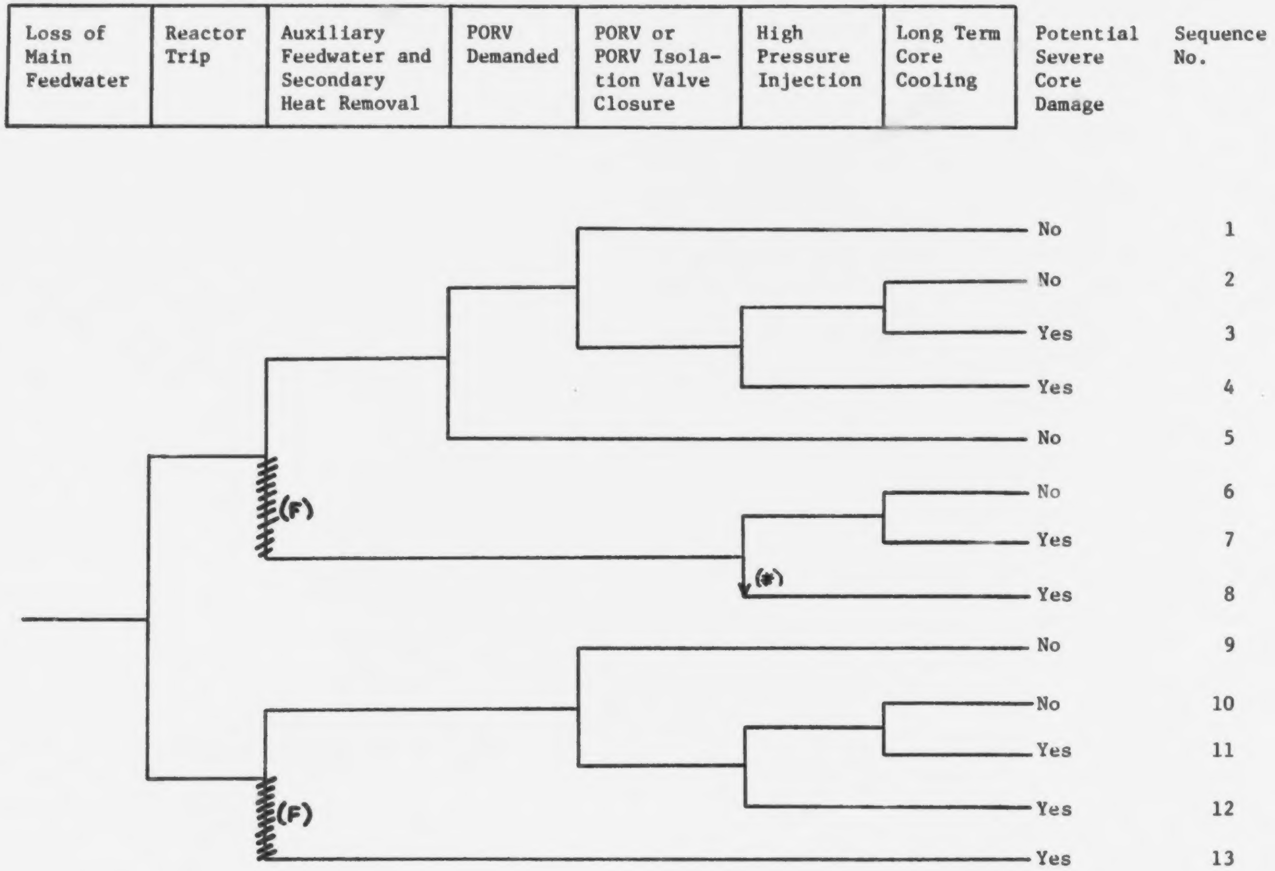
Unavailability of component per WASH 1400:* strainers, not included
pumps, failure to run: 3×10^{-5}/hr.

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Unit 1 in cooldown
Unit 2 status unknown
Motor-driven AFW pump A fails to deliver adequate flow to Unit 1 due to clogged suction strainer
Motor-driven AFW pump B suction strainer clogged
Turbine-driven AFW pump suction strainer partially clogged (applicable to both unit turbine-driven pumps)
Potential Severe Core Damage
Yes
No
No — offsite power for main feed pumps available
No
No
No

NSIC 90421 — Actual Occurrence for Auxiliary Feed Pump Suction Strainers Clogged at Point Beach 1 & 2


Loss of Main Feedwater
Reactor Trip
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
Yes 3
Yes 4
No 5
No 6
Yes 7
Yes 8
No 9
No 10
Yes 11
Yes 12
Yes 13
(F)
(F)
(*)

NSIC 90421 — Sequence of Interest for Auxiliary Feed Pump Suction Strainers Clogged at Point Beach 1 & 2
*Use of HPI following AFW failure not included in mitigation procedures.

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 90421

DATE OF LER: April 17, 1974

DATE OF EVENT: April 7, 1974

SYSTEM INVOLVED: Auxiliary feedwater

COMPONENT INVOLVED: AFW pump strainers

CAUSE: Strainers plugged

SEQUENCE OF INTEREST: Loss of feedwater

ACTUAL OCCURRENCE: Inoperable AFW pumps during plant cooldown.

REACTOR NAME: Point Beach 1

DOCKET NUMBER: 50-266

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 497 MWe

REACTOR AGE: 3.5 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Wisconsin Electric Power Co.

LOCATION: **15 miles north of Manitowoc, Wisc.**

DURATION: 360(a) hours

PLANT OPERATING CONDITION: Unit 1 in cooldown

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ___________________

DISCOVERY METHOD: during operation

COMMENT: The affected AFW pumps provide auxiliary feedwater for both units. Monthly tests failed to reveal strainers because test was run at shutoff head — the only flow was through a 1" ID bypass line & this low flow masked the plugged strainer.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 91676

Date: May 17, 1974

Title: Malfunction of Auxiliary Feedwater Pumps at Turkey Point 3

The failure sequence was:

1. With the reactor at power, an attempt was made to start all three auxiliary feedwater pumps.
2. Two of the pumps failed to start due to overtightened packings.
3. The third pump failed to start due to a turbine regulating valve pneumatic controller malfunction.

Corrective action;

1. The packing on the "A" AFW pump was loosened and later replaced. After replacement, a sticking governor linkage was also discovered.
2. The "B" AFW pump was repacked.
3. The "C" AFW pump governor was exercised and its pneumatic controller output signal was increased in value.

Design purpose of failed system or component:

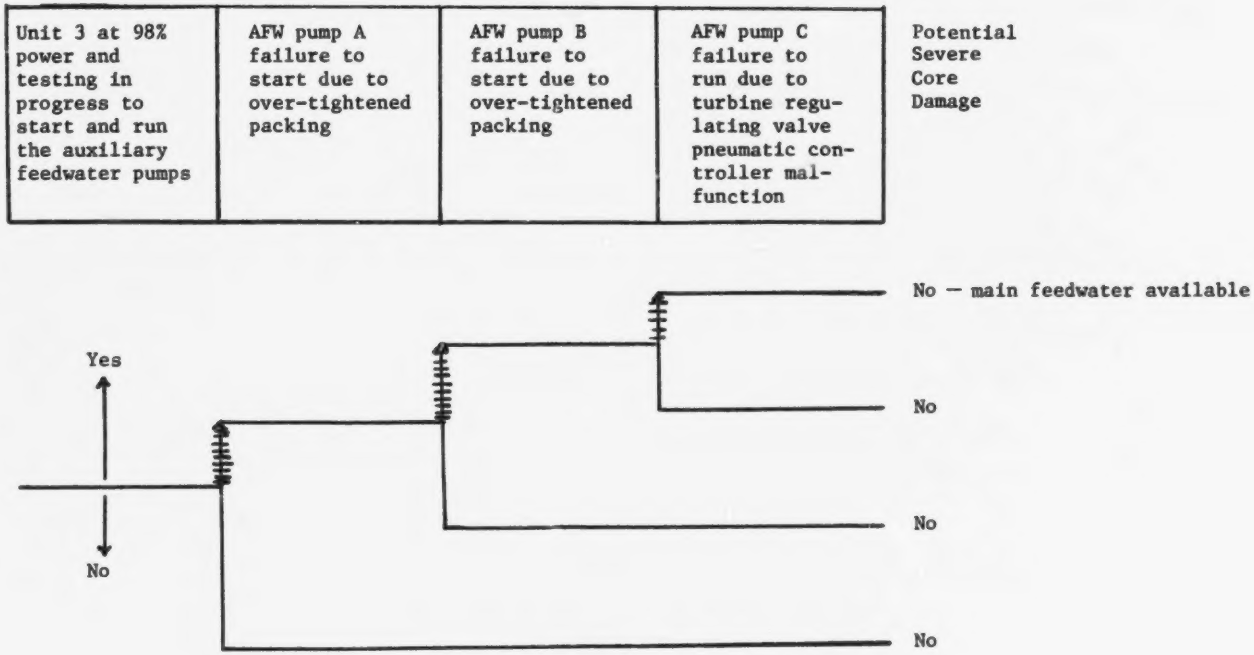
1. The auxiliary feedwater system provides feedwater to the steam generators for reactor cooling when main feedwater is unavailable.

Unavailability of system per WASH 1400:* Auxiliary feedwater system: 2.5 × 10^{-4} (start + 8 hours)

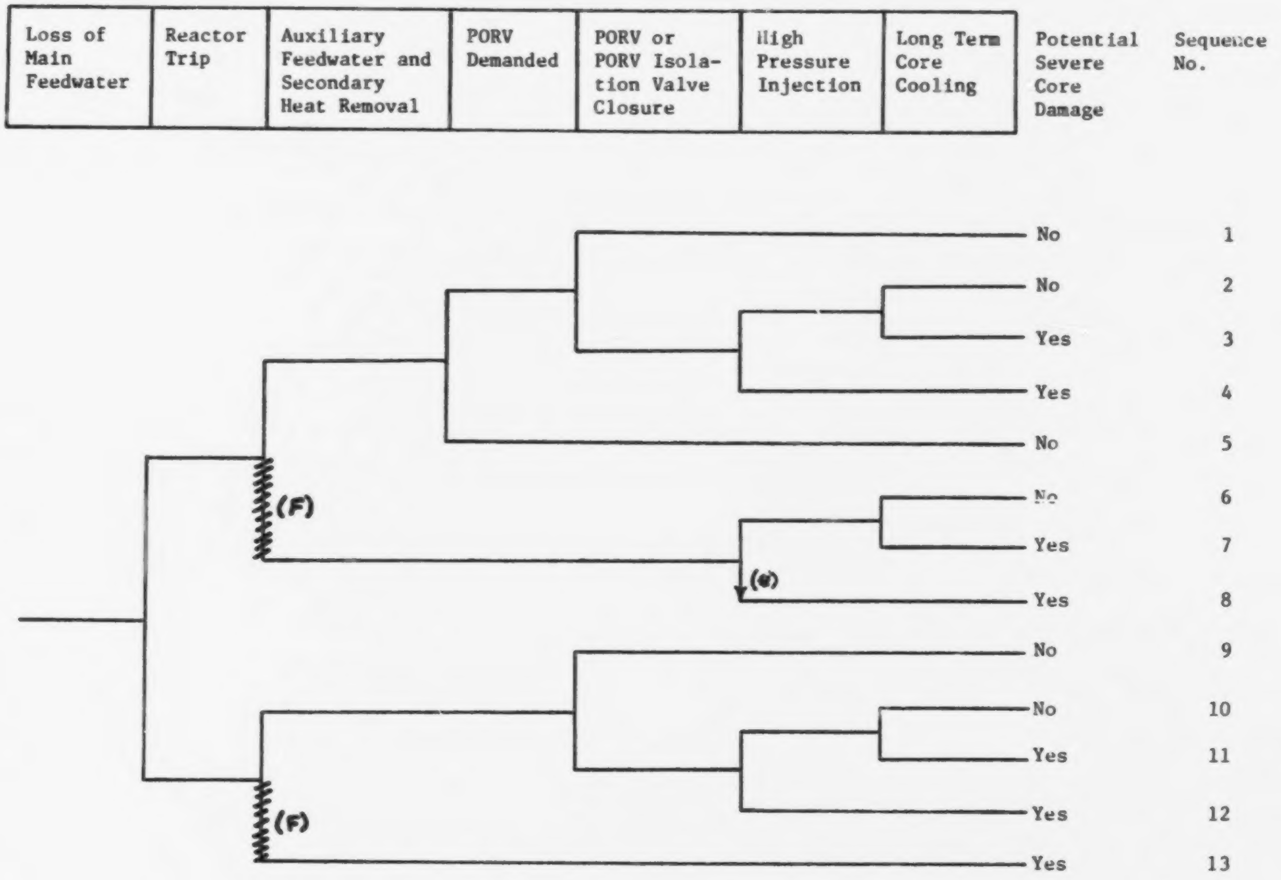
Unavailability of component per WASH 1400:* pump, failure to start: 1 × 10^{-3}/D

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Unit 3 at 98% power and testing in progress to start and run the auxiliary feedwater pumps
AFW pump A failure to start due to over-tightened packing
AFW pump B failure to start due to over-tightened packing
AFW pump C failure to run due to turbine regulating valve pneumatic controller malfunction
Potential Severe Core Damage
Yes
No
No — main feedwater available
No
No
No

NSIC 91676 — Actual Occurrence for Malfunction of Auxiliary Feedwater Pumps at Turkey Point 3


Loss of Main Feedwater
Reactor Trip
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
Yes 3
Yes 4
No 5
No 6
Yes 7
Yes 8
No 9
No 10
Yes 11
Yes 12
Yes 13
(F)
(*)
(F)

NSIC 91676 — Sequence of Interest for Malfunction of Auxiliary Feedwater Pumps at Turkey Point 3
*Use of HPI following AFW failure not included in mitigation procedures.

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 91676

DATE OF LER: May 17, 1974

DATE OF EVENT: May 8, 1974

SYSTEM INVOLVED: Auxiliary feedwater

COMPONENT INVOLVED: Auxiliary feedwater pumps

CAUSE: Overtightened packings, turbine regulating valve pneumatic controller malfunction, human maintenance error

SEQUENCE OF INTEREST: Loss of feedwater

ACTUAL OCCURRENCE: Failure of three auxiliary feedwater pumps to start and run during testing.

REACTOR NAME: Turkey Point 3

DOCKET NUMBER: 50-250

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 693 MWe

REACTOR AGE: 1.9 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Florida Power & Light Co.

LOCATION: 25 miles south of Miami, Fla.

DURATION: 360(a) hours

PLANT OPERATING CONDITION: At 98% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; [✓ (b)] failed to start; (c) made inoperable; (d) ________________

DISCOVERY METHOD: During testing

COMMENT: The three AFW pumps which failed to start also provide auxiliary feedwater to Unit 4.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 93553

Date: December 8, 1974

Title: Diesel Generator Malfunctions at Hatch 1

The failure sequence was:

1. The diesel generators were undergoing surveillance testing.
2. Diesel generator 1A failed to start.
3. Diesel generator 1B started but shut down after about a minute of operation.
4. An investigation revealed that the air side of the governor booster had rusted on both diesels. Also the day tank for diesel 1G contained 50 gallons of water.

Corrective action:

The air port to the booster was cleaned and its diameter increased from 0.025 to 0.050 inches on all three diesels.

Design purpose of failed system or component:

The diesels provide a.c. power to the plant when offsite sources are unavailable.

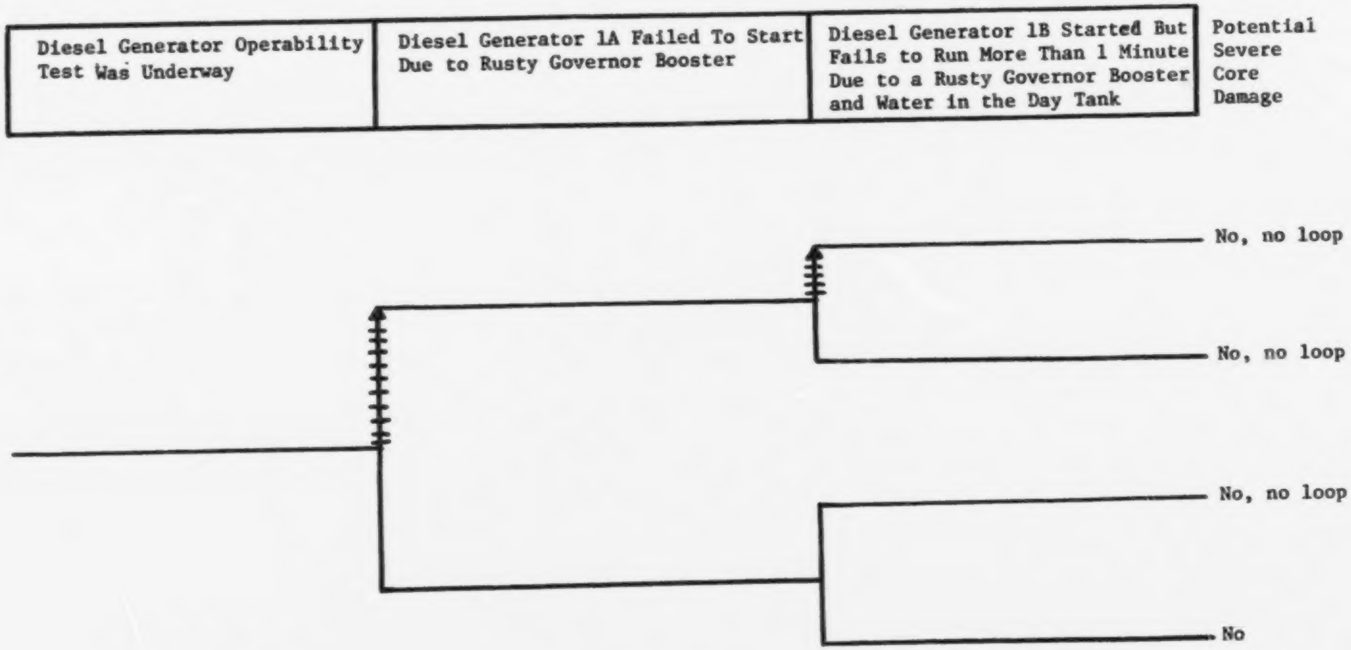
The governor booster supplies oil to the governor.

Unavailability of system per WASH 1400:* Emergency Power: 10^{-2}/D

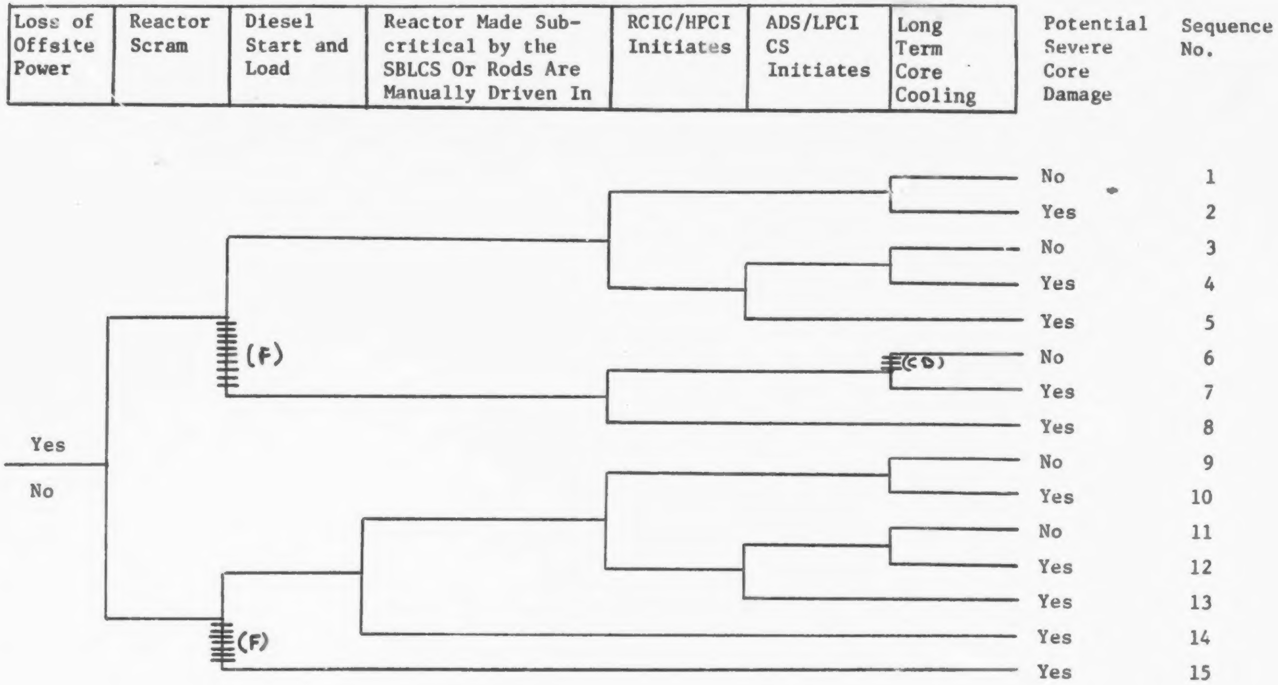
Unavailability of component per WASH 1400:* Diesel failure to start: 3×10^{-2}/D
Diesel Engine failure to run:
3×10^{-4}/hr

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Diesel Generator Operability Test Was Underway
Diesel Generator 1A Failed To Start Due to Rusty Governor Booster
Diesel Generator 1B Started But Fails to Run More Than 1 Minute Due to a Rusty Governor Booster and Water in the Day Tank
Potential Severe Core Damage
No, no loop
No, no loop
No, no loop
No

NSIC 93553 — Actual Occurrence of Two Diesels Fail at Hatch 1


Loss of Offsite Power
Reactor Scram
Diesel Start and Load
Reactor Made Subcritical by the SBLCS Or Rods Are Manually Driven In
RCIC/HPCI Initiates
ADS/LPCI CS Initiates
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
(F)
(CD)
(F)
No 1
Yes 2
No 3
Yes 4
Yes 5
No 6
Yes 7
Yes 8
No 9
Yes 10
No 11
Yes 12
Yes 13
Yes 14
Yes 15

NSIC 93553 — Sequence of Interest of Two Diesel Generators Fail at Hatch 1

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 93553

DATE OF LER: January 16, 1975

DATE OF EVENT: December 8, 1974

SYSTEM INVOLVED: Electric Power

COMPONENT INVOLVED: Diesel Generators

CAUSE: Mechanical failure of the governor boosters, water in the day tank

SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE: Two diesel generators malfunction at Hatch 1

REACTOR NAME: Hatch 1

DOCKET NUMBER: 50-321

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 786 MWe

REACTOR AGE: yr

VENDOR: General Electric

ARCHITECT-ENGINEERS:

OPERATORS: Georgia Power

LOCATION: 11 miles North of Baxley, GA

DURATION: 360(a) hours

PLANT OPERATING CONDITION: not known

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; [✓ (b)] failed to start; (c) made inoperable; [✓ (d)] failed to run

DISCOVERY METHOD: Testing

COMMENT: Diesel 1A had failed on December 7.

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 93702

Date: May 31, 1974

Title: Two-Unit Trip at Turkey Point 3 and 4, Loss of Offsite Power at Turkey Point 4

The failure sequence was:

1. Units 3 & 4 were at full power, with unit 4 startup transformer electrically isolated for functional testing of protection system relays and for periodic maintenance on associated equipment.
2. After the transformer protective relays were tested, the transformer lockout relay was reset and current flowed through an unanticipated path (a "sneak circuit") and energized unit 4 generator breaker protection system relays, which opened the generator breakers for units 3 and 4.
3. Both unit 3 & 4 reactors automatically tripped when their turbine-generators tripped. The instantaneous grid loss was 1443 MWe.
4. Since the unit 4 startup transformer was isolated, off-site power was lost to unit 4.

(see additional page)

Corrective action:

1. Modifications were recommended to the startup transformer protective system to provide more complete isolation of the protection system relays during relay tests.
2. The stuck-open safety valve was closed by removing its manual lifting-device.

Design purpose of failed system or component:

1. Offsite electric power provides an alternative source of power to plant auxiliaries in the event the unit generator is inoperable.
2. The main steam safety valves provide over-pressure protection for the steam generators and main-steam piping.

Unavailability of system per WASH 1400:* Offsite power: 10^{-3} following trip

Unavailability of component per WASH 1400:* Relief valves, failure to close: 10^{-2}/D

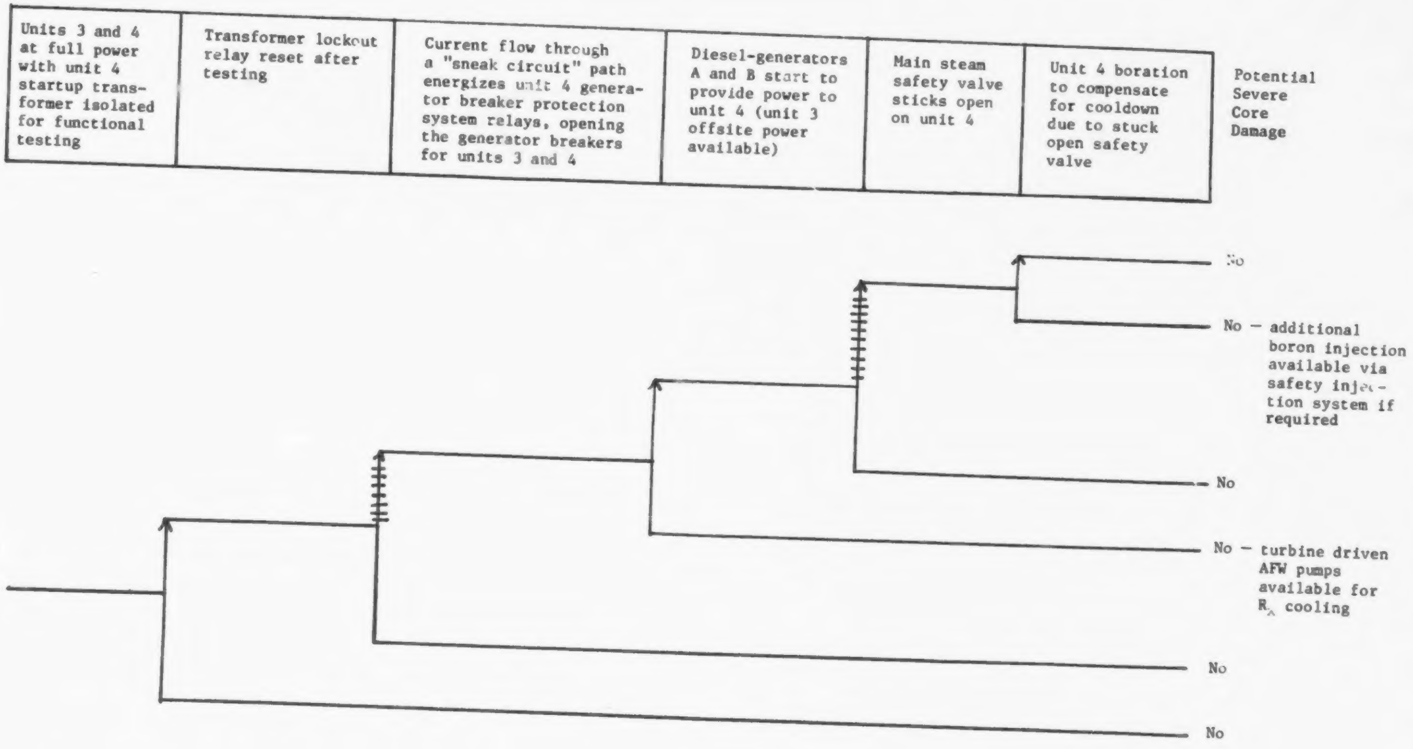
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

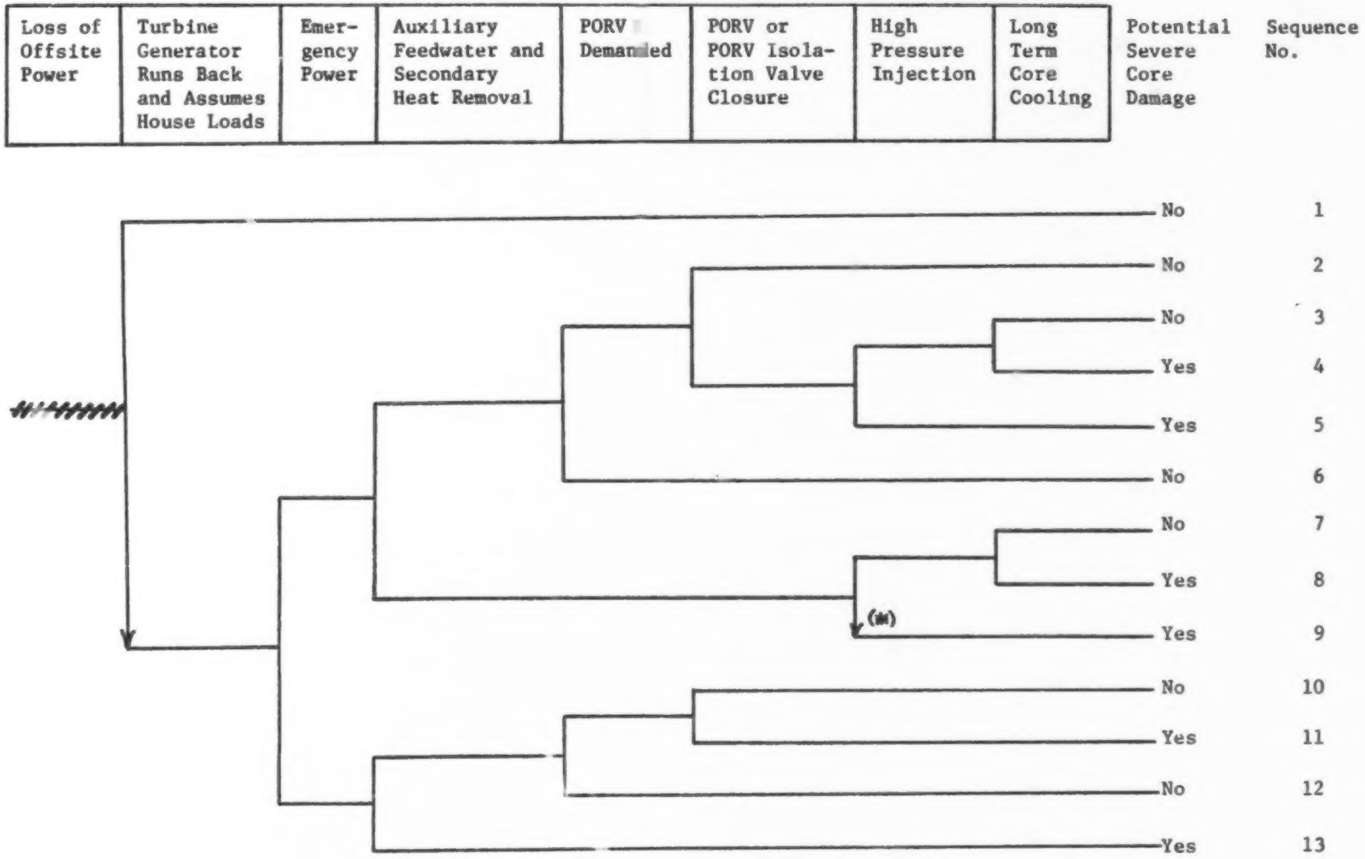
[redacted]

The failure sequence was: (Continued)

5. Diesel-generators A & B started and assumed safety-related loads for unit 4.
6. Five minutes after reactor trip on unit 4, a main steam safety valve lifted and did not reseat.
7. Additional boron was added to the reactor coolant system to provide shutdown margin for the additional cooldown resulting from the stuck-open safety valve.


Units 3 and 4 at full power with unit 4 startup transformer isolated for functional testing
Transformer lockout relay reset after testing
Current flow through a "sneak circuit" path energizes unit 4 generator breaker protection system relays, opening the generator breakers for units 3 and 4
Diesel-generators A and B start to provide power to unit 4 (unit 3 offsite power available)
Main steam safety valve sticks open on unit 4
Unit 4 boration to compensate for cooldown due to stuck open safety valve
Potential Severe Core Damage
No
No — additional boron injection available via safety injection system if required
No
No — turbine driven AFW pumps available for R_x cooling
No
No

NSIC 93702 — Actual Occurrence for Two-Unit Trip at Turkey Point 3 & 4


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
(*)
Yes 9
No 10
Yes 11
No 12
Yes 13

NSIC 93702 — Sequence of Interest for Two Unit Trip at Turkey Point 3 & 4
*Use of HPI following AFW failure not included in mitigation procedures.

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 93702

DATE OF LER: May 31, 1974

DATE OF EVENT: April 25, 1974

SYSTEM INVOLVED: Electric power system, main steam

COMPONENT INVOLVED: Startup transformer protective relays, main steam safety valve

CAUSE: Current flow in a sneak circuit during protective relay testing resulted in actuation of the Unit 3 & 4 generator breakers. After reactor trip, a Unit 4 main steam safety valve stuck open.

SEQUENCE OF INTEREST: Reactor trip with loss of offsite power

ACTUAL OCCURRENCE: Reactor trip with loss of offsite power (Unit 4 only)

REACTOR NAME: Turkey Point 4

DOCKET NUMBER: 50-251

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 693 MWe

REACTOR AGE: .9 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Florida Power & Light Co.

LOCATION: 25 miles south of Miami, Fla.

DURATION: N/A

PLANT OPERATING CONDITION: Both units at full power.

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; (c) made inoperable; [✓ (d) inadvertant operation]

DISCOVERY METHOD: During operation

COMMENT: This event and a previous event on March 1, 1974 resulted in the separation of both Turkey Point Units from the grid. However, only Unit #4 and only during this event lost offsite power.

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 95040

Date:

Title: HPCI Injection Valves and RCIC Fail at Browns Ferry 2

The failure sequence was:

1. The HPCI injection valve failed to open during post-maintenance testing of this valve.
2. RCIC was then initiated, however, it failed to show any speed indication. The seat holding the valve operative was broken and a printed circuit board in the tachometer was damaged.

Corrective action:

1. The broken valve operator seats were replaced.
2. A new circuit board was installed in the RCIC tachometer.

Design purpose of failed system or component:

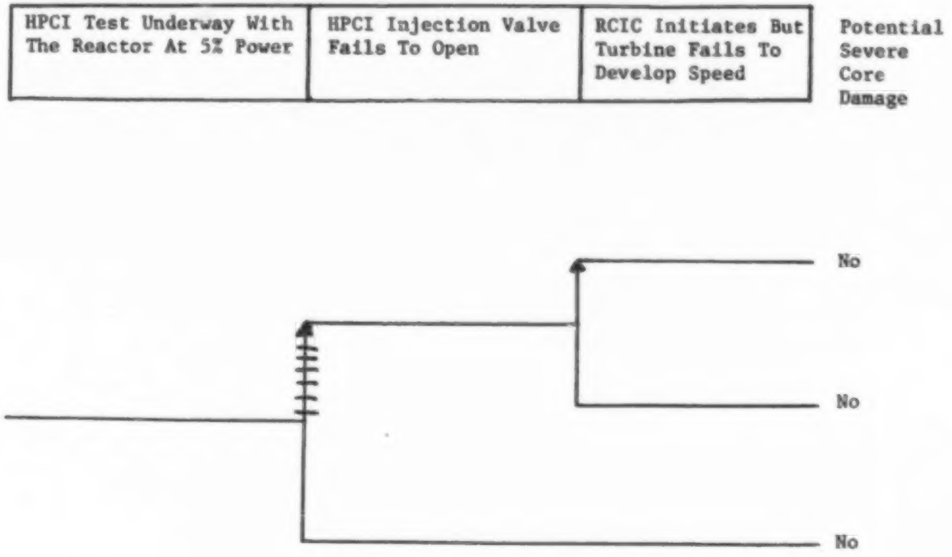
1. RCIC provides a source of water should the reactor become isolated.
2. HPCI is a source of high pressure cooling water given a small LOCA.

Unavailability of system per WASH 1400:* RCIC: 8.0×10^{-2}/D
HPCI: 8.8×10^{-2}/D

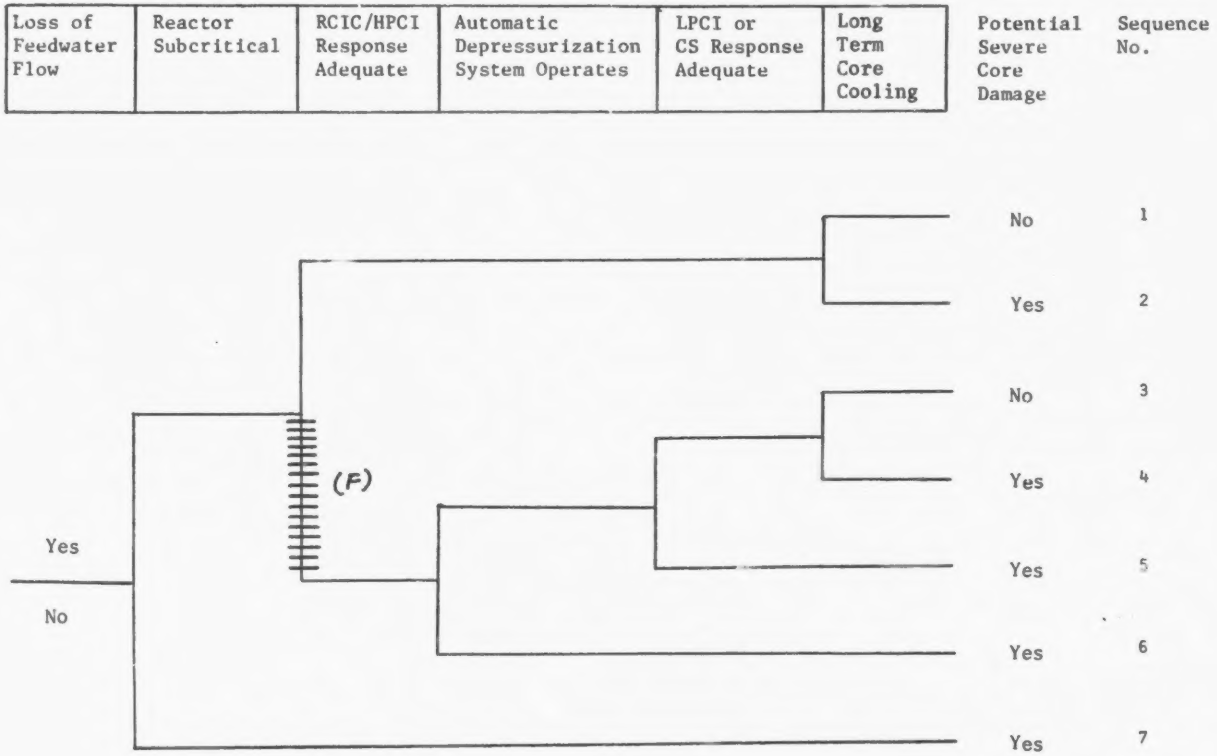
Unavailability of component per WASH 1400:* MOV *failure to operate: 1.0×10^{-3}/D

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


HPCI Test Underway With The Reactor At 5% Power
HPCI Injection Valve Fails To Open
RCIC Initiates But Turbine Fails To Develop Speed
Potential Severe Core Damage
No
No
No

NSIC 95040 — Actual Occurrence of HPCI Injection Valves and RCIC Fail at Browns Ferry 2


Loss of Feedwater Flow
Reactor Subcritical
RCIC/HPCI Response Adequate
Automatic Depressurization System Operates
LPCI or CS Response Adequate
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
(F)
No 1
Yes 2
No 3
Yes 4
Yes 5
Yes 6
Yes 7

NSIC 95040 — Sequence of Interest for HPCI Injection Valves and RCIC Fail at Browns Ferry 2

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 95040

DATE OF LER: August 23, 1974

DATE OF EVENT:

SYSTEM INVOLVED: RCFC, HPCI

COMPONENT INVOLVED: Valves, tachometer circuit boards.

CAUSE: Mechanical Failure

SEQUENCE OF INTEREST: Loss of Feedwater

ACTUAL OCCURRENCE: HPCI Injection Valves and RCIC Fail at Browns Ferry 2

REACTOR NAME: Browns Ferry 2

DOCKET NUMBER: 260

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 1065 MWe

REACTOR AGE: .09 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: TVA

OPERATORS: TVA

LOCATION: 10 miles NW of Decatur, AL

DURATION: 360(a) hours

PLANT OPERATING CONDITION: Operating at 5% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________________

DISCOVERY METHOD: Operational Event.

COMMENT: No LER

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 97107

Date: October 31, 1974

Title: BIT Auto-Injection Failure at Zion 2

Meets precursor criteria no:

The failure sequence was:

1. An instrument mechanic inadvertently tripped a RCS pump.
2. The resulting flow transient caused a reactor trip and SI signal.
3. Train A safeguards failed due to a loss of power.
4. The train B inlet valve to BIT failed to open.

Corrective action:

Valve stem was cleaned, packing was changed, and torque limits were increased.

Design purpose of failed system or component:

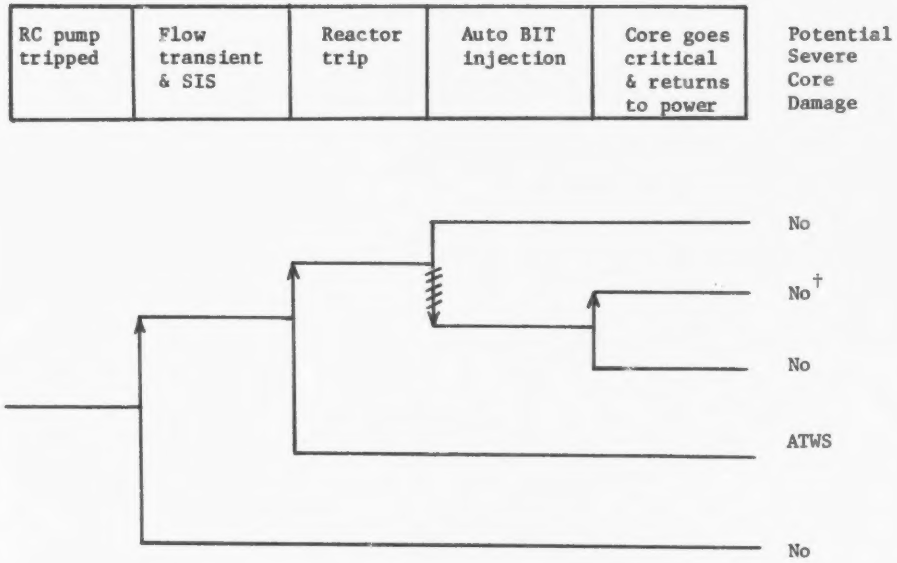
BIT protects the core from a main steam pipe break.

Unavailability of system per WASH 1400:* -

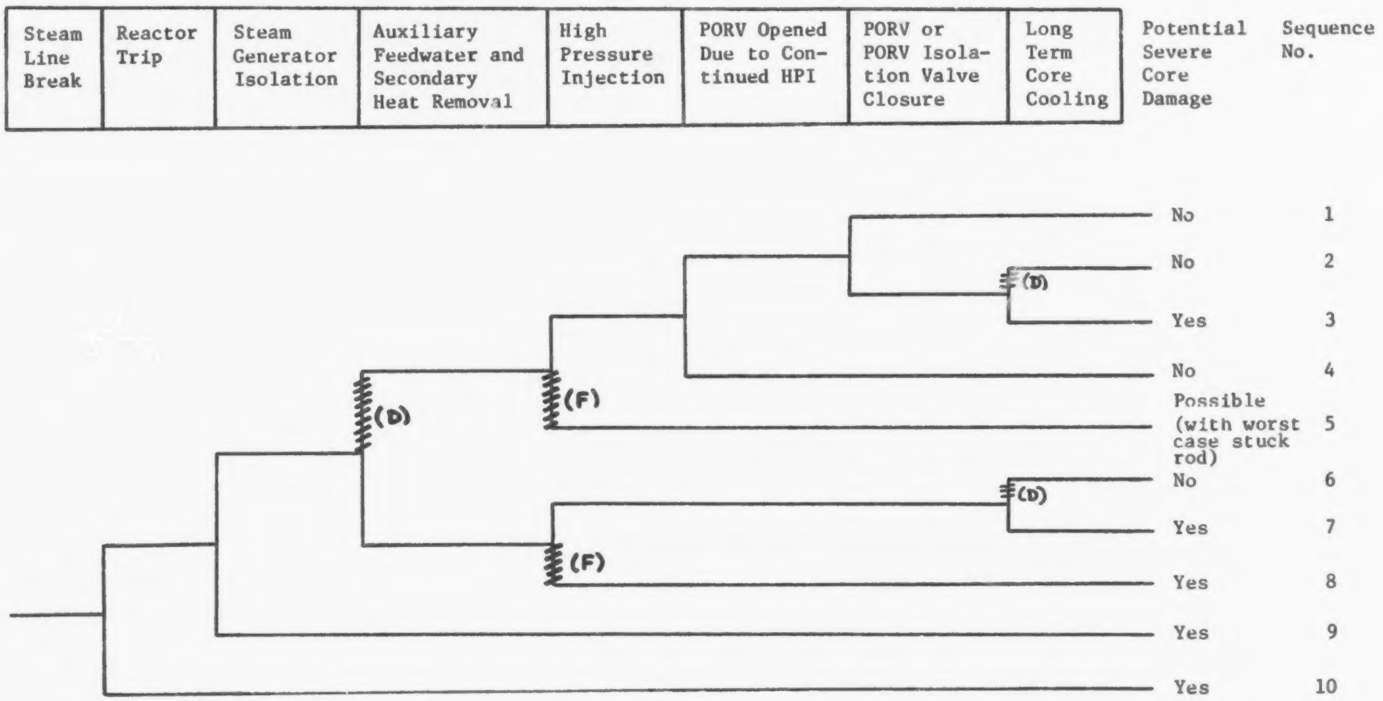
Unavailability of component per WASH 1400:* valve, failure to operate: 1×10^{-3}/D
safeguards, train A: 5.8×10^{-3}/D
125 VDC system: 1.1×10^{-6}/D

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


RC pump tripped
Flow transient & SIS
Reactor trip
Auto BIT injection
Core goes critical & returns to power
Potential Severe Core Damage
No
No†
No
ATWS
No

NSIC 97107 — Actual Occurrence for Boron Injection Tank Inlet Valve Failures at Zion 2 (BIT is primarily for main steam pipe break — FSAR says no core melt for any pump trips. †To return to criticality would require the most reactive RCCA to be stuck fully withdrawn and end of core life negative coolant temperature coefficient.)


Steam Line Break
Reactor Trip
Steam Generator Isolation
Auxiliary Feedwater and Secondary Heat Removal
High Pressure Injection
PORV Opened Due to Continued HPI
PORV or PORV Isolation Valve Closure
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
(D)
(F)
(D)
(F)
(D)
No 1
No 2
Yes 3
No 4
Possible (with worst case stuck rod) 5
No 6
Yes 7
Yes 8
Yes 9
Yes 10

NSIC 97107 — Sequence of Interest for Boron Injection Tank Valve Failures at Zion 2

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 97107

DATE OF LER: October 31, 1974

DATE OF EVENT: October 22, 1974

SYSTEM INVOLVED: Safety Injection

COMPONENT INVOLVED: Valves — 2

CAUSE: Power failure to one safeguards train and stuck valve on other train, (human error)

SEQUENCE OF INTEREST: Main steam pipe break

ACTUAL OCCURRENCE: Instrument mechanic inadvertently tripped a RCS pump, operator error

REACTOR NAME: Zion 2

DOCKET NUMBER: 50-304

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 1040 MWe

REACTOR AGE: 0.8 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Sargent & Lundy

OPERATORS: Commonwealth Edison Co.

LOCATION: 40 miles north of Chicago, Ill.

DURATION: 360(a) hours

PLANT OPERATING CONDITION: 57% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ______________

DISCOVERY METHOD: Operational event

COMMENT:

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 97578

Date: November 15, 1974

Title: Loss of Offsite Power During SIS Testing at Palisades

The failure sequence was:

1. With the plant in a hot standby condition, the right channel safety injection test button was pushed to initiate a quarterly test.
2. Offsite power was lost due to the inadvertent operation of the differential relay system.
3. The diesel generators started and powered safety-related loads.

Corrective action:

The three-phase differential relays were removed from service pending a review and potential redesign of the system. Over-current protection devices remained installed to provide transformer protection.

Design purpose of failed system or component:

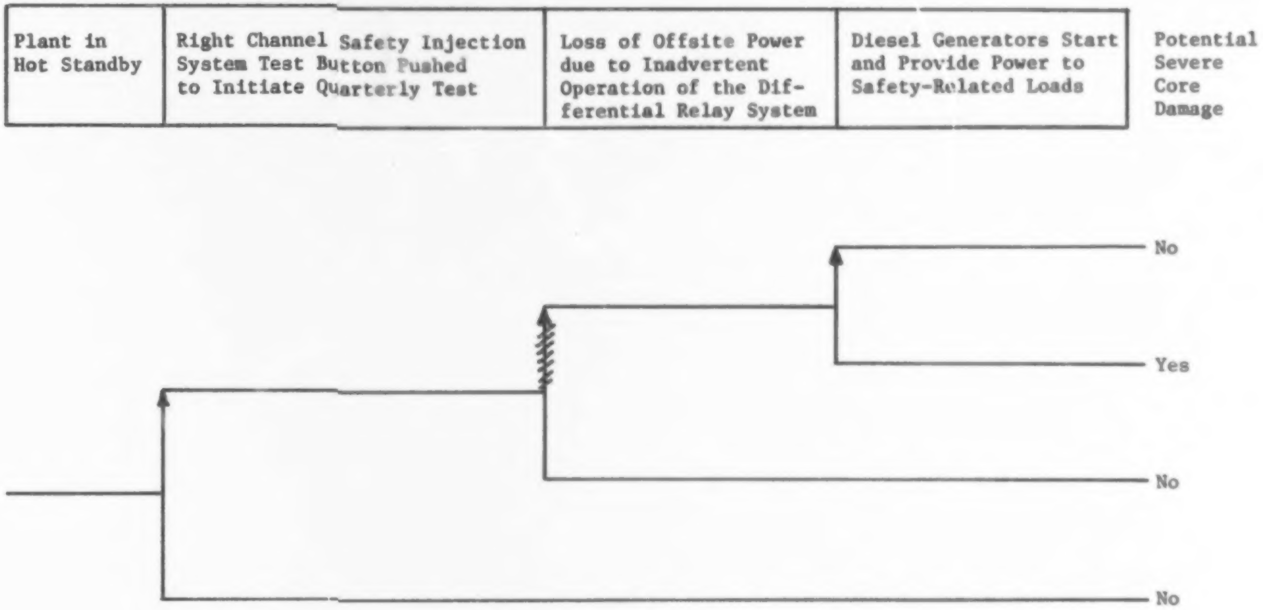
The differential relays provided overcurrent protection for the startup transformers.

Unavailability of system per WASH 1400:* Offsite power: 2×10^{-5}/hr

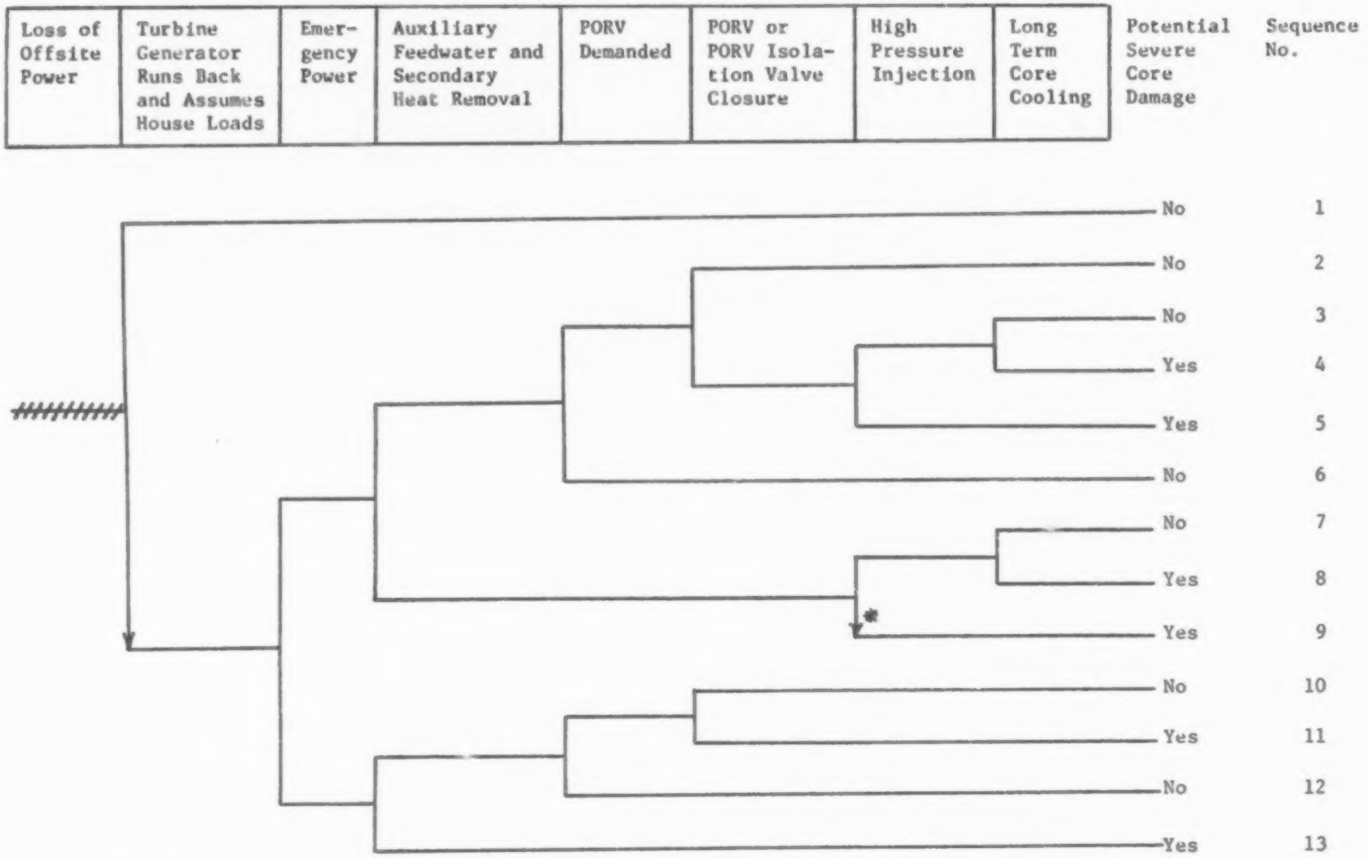
Unavailability of component per WASH 1400:* -

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Plant in Hot Standby
Right Channel System Test Button Pushed to Initiate Quarterly Test
Safety Injection
Loss of Offsite Power due to Inadvertent Operation of the Differential Relay System
Diesel Generators Start and Provide Power to Safety-Related Loads
Potential Severe Core Damage
No
Yes
No
No

NSIC 97578 — Actual Occurrence for Loss of Offsite Power During SIS Testing at Palisades


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
*
Yes 9
No 10
Yes 11
No 12
Yes 13

NSIC 97578 — Sequence of Interest for Loss of Offsite Power During SIS Testing at Palisades

*Not included in mitigation sequence.

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 97578

DATE OF LER: November 15, 1974

DATE OF EVENT: October 17, 1974

SYSTEM INVOLVED: Offsite Power

COMPONENT INVOLVED: Differential Relays

CAUSE: Improperly designed Differential Relay System

SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE: Loss of offsite power during SIS testing

REACTOR NAME: Palisades

DOCKET NUMBER: 50-255

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 805 MWe

REACTOR AGE: 3.4 yr

VENDOR: Combustion Engineering

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Consumers Power Company

LOCATION: 5 miles south of South Haven, Mich.

DURATION: N/A

PLANT OPERATING CONDITION: hot standby

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________________

DISCOVERY METHOD: testing

COMMENT: This event is the second of its type at Palisades. After the first event (NSIC 71694, 5/17/72), the differential relays were removed pending a design review. They were reinstalled in January 1974 after modification.

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 97779

Date: December 3, 1974

Title: Reactor Coolant Pump Underpower/Overpower Monitoring Relays Fail to Function at Arkansas Nuclear 1

The failure sequence was:

1. With the reactor at 20% power, RCPs A and D were shutdown for two pump testing. The RPS saw neither pump trip.
2. RCPs A and D were restarted and RCPs B and C were tripped. The RPS saw neither pump trip.
3. Subsequent testing indicated the RPS failed to see a RCP D trip but did see a RCP A trip.
4. Concurrent with a trip of the C RCP, the D pump monitor relay also tripped (D pump still running), resulting in a reactor trip.

(See attached page)

Corrective action:

The relays were correctly readjusted for the pump power monitor application.

Design purpose of failed system or component:

The pump power monitors provide pump operating status indication to the RPS. The RPS is designed to trip the reactor if more than one pump is lost during four pump operation or if any pump is lost during less than four loop operation.

Unavailability of system per WASH 1400:* Not considered.

Unavailability of component per WASH 1400:* Relay, failure to de-energize: not specified.

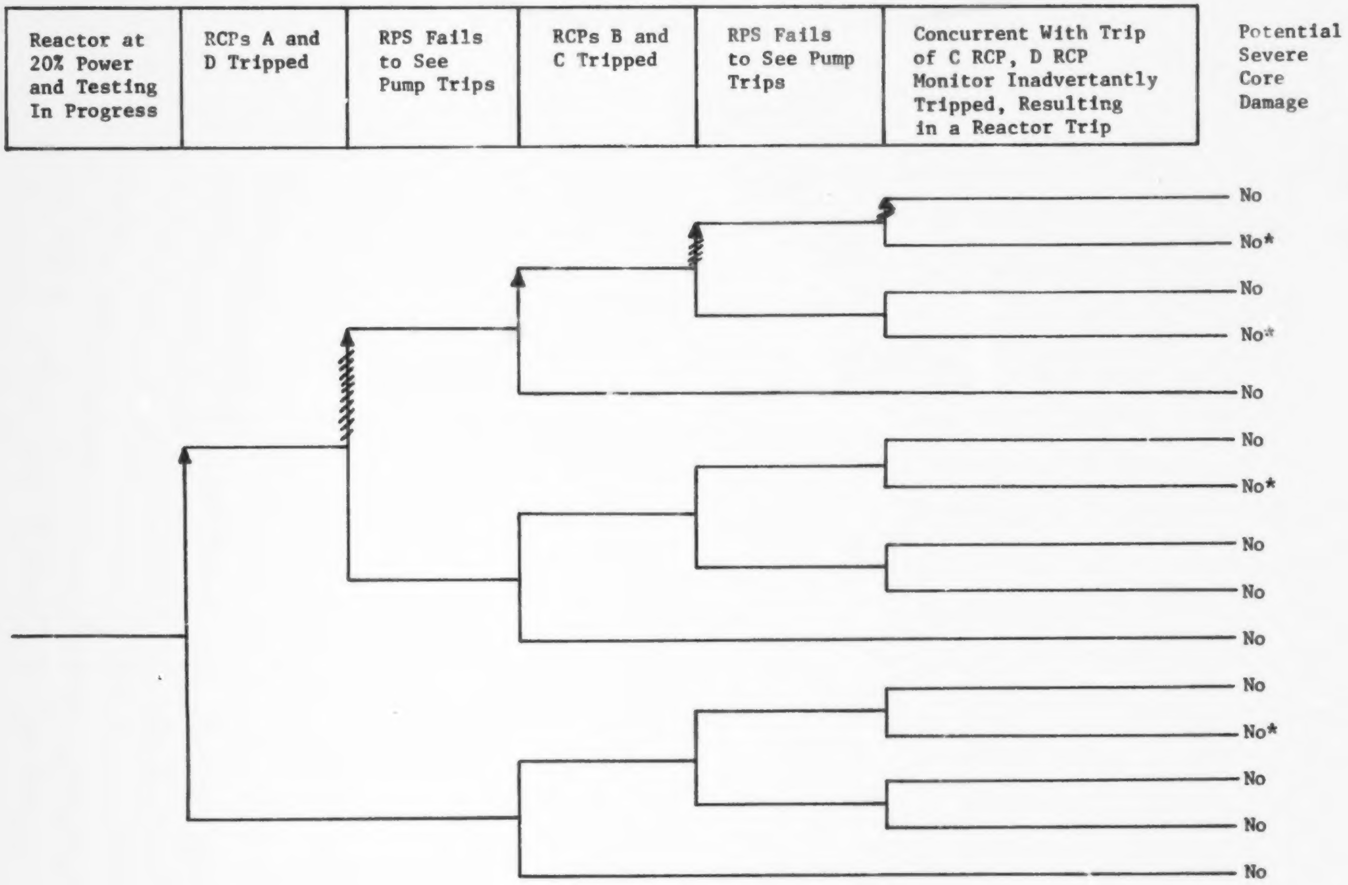
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

[redacted]

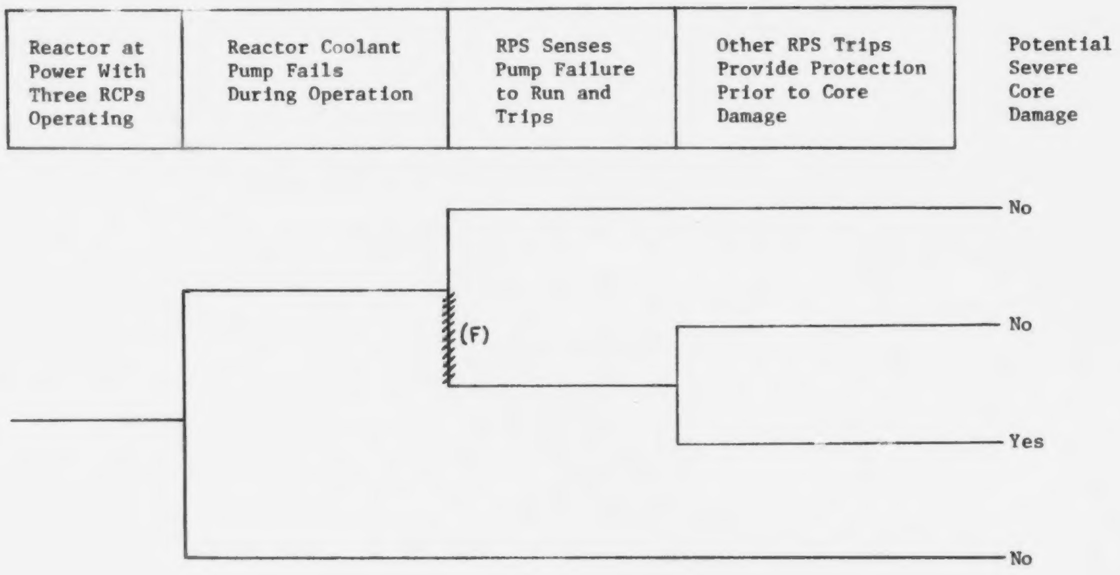
The failure sequence was: (continued)

5. Investigation revealed that the relays had been adjusted two days before in accordance with the relay manufacturers instructions. However, in this application, the relays had to be adjusted to operate at a specific pre-determined power level.


Reactor at 20% Power and Testing In Progress
RCPs A and D Tripped
RPS Fails to See Pump Trips
RCPs B and C Tripped
RPS Fails to See Pump Trips
Concurrent With Trip of C RCP, D RCP Monitor Inadvertantly Tripped, Resulting in a Reactor Trip
Potential Severe Core Damage
No
No*
No
No*
No
No
No*
No
No
No
No
No*
No
No
No

NSIC 97779 — Actual Occurrence for Reactor Coolant Pump Underpower/Overpower Monitoring Relays Fail to Function at Arkansas 1

*No transient requiring RCP trips existed


Reactor at Power With Three RCPs Operating
Reactor Coolant Pump Fails During Operation
RPS Senses Pump Failure to Run and Trips
Other RPS Trips Provide Protection Prior to Core Damage
Potential Severe Core Damage
No
(F)
No
Yes
No

NSIC 97779 — Sequence of Interest for Reactor Coolant Pump Undervoltage/Overpower Monitoring Relays Fail to Function at Arkansas Nuclear 1

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 97779

DATE OF LER: December 3, 1974

DATE OF EVENT: November 24, 1974

SYSTEM INVOLVED: Reactor Protection System

COMPONENT INVOLVED: Pump Power Monitor Relays

CAUSE: Relays incorrectly adjusted

SEQUENCE OF INTEREST: Reactor coolant pump trip during three loop operation

ACTUAL OCCURRENCE: Failure of RPS to see trips during testing

REACTOR NAME: Arkansas Nuclear 1

DOCKET NUMBER: 50-313

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 850 MWe

REACTOR AGE: .30 yr

VENDOR: B&W

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Arkansas Power & Light Co.

LOCATION: **6 miles NW of Russelville, Ark.**

DURATION: 48 hours

PLANT OPERATING CONDITION: 20% power

SAFETY FEATURE TYPE OF FAILURE: [✓ (a)] inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________________

DISCOVERY METHOD: During testing

COMMENT:

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 100415

Date: March 5, 1975

Title: All Drywell High Pressure Switches Found Isolated at Fitzpatrick

The failure sequence was:

During routine surveillance testing, an instrumentation technician discovered the rack isolation valves for drywell high pressure instruments in both trains closed. Further investigation revealed the associated root valves at the drywell penetrations were also closed. It is presumed the valves were left closed since August 1974 when the valves were supposed to be reopened following the completion of surveillance testing.

Corrective action:

1. The valves were immediately opened and locked to prevent reoccurrence of this event.
2. The root valves and other isolation valves are being added to appropriate operational and maintenance check lists.

Design purpose of failed system or component:

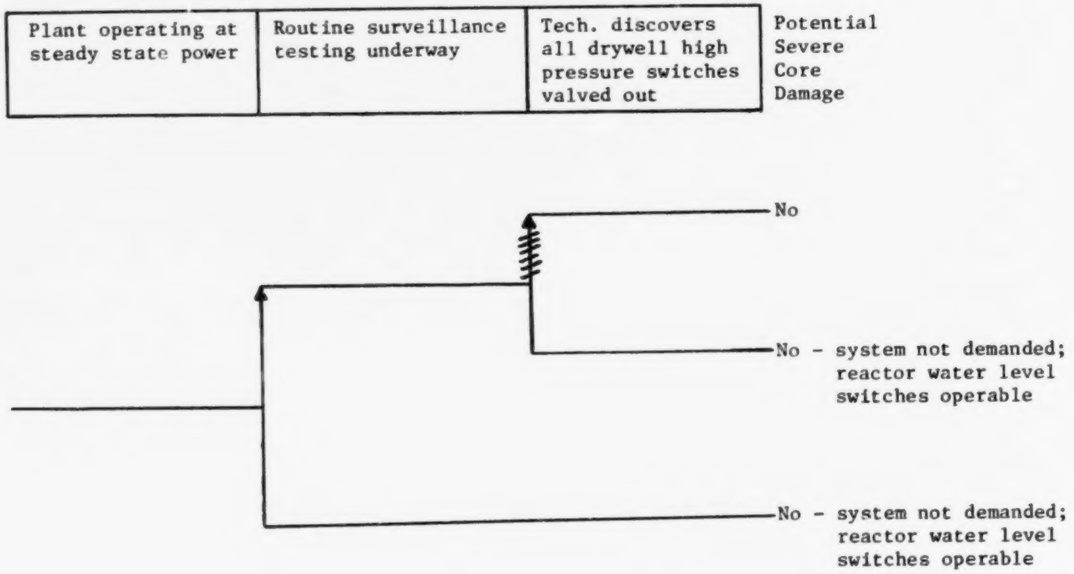
The high drywell pressure switches send initiation on signals to the RPS & ECCS Initiation Systems.

Unavailability of system per WASH 1400:* -

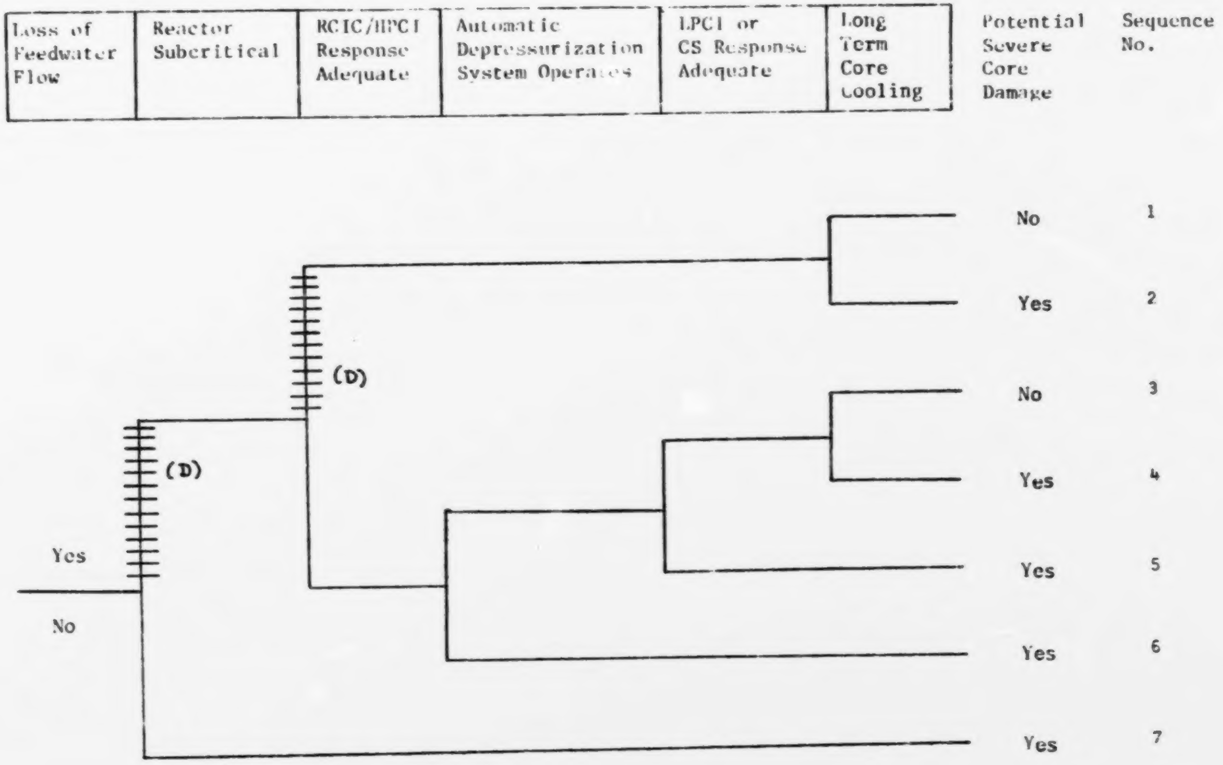
Unavailability of component per WASH 1400:* 3×10^{-3}/D general human error

---

* Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Plant operating at steady state power
Routine surveillance testing underway
Tech. discovers all drywell high pressure switches valved out
Potential Severe Core Damage
No
No - system not demanded; reactor water level switches operable
No - system not demanded; reactor water level switches operable

NSIC 100415 — Actual Occurrence of All High Pressure Switches Found Isolated at FitzPatrick


Loss of Feedwater Flow
Reactor Subcritical
RCIC/HPCI Response Adequate
Automatic Depressurization System Operates
LPCI or CS Response Adequate
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
(D)
(D)
No 1
Yes 2
No 3
Yes 4
Yes 5
Yes 6
Yes 7

NSIC 100415 - Sequence of Interest for All Drywell Pressure Switches Found Isolated at Fitzpatrick

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 100415

DATE OF LER: March 5, 1975

DATE OF EVENT: February 25, 1975

SYSTEM INVOLVED: reactor protection, emergency core cooling system

COMPONENT INVOLVED: drywell high pressure switch

CAUSE: human error

SEQUENCE OF INTEREST: loss of feedwater

ACTUAL OCCURRENCE: all drywell high pressure switches found isolated at Fitzpatrick

REACTOR NAME: Fitzpatrick

DOCKET NUMBER: 50-333

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 821 MWe

REACTOR AGE: .3 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: Stone & Webster

OPERATORS: Power Authority of the state of New York

LOCATION: 8 miles NE of Oswego, NY

DURATION: 5110 hours

PLANT OPERATING CONDITION: steady state operating conditions

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c) made inoperable]; (d) ______________________

DISCOVERY METHOD: surveillance testing

COMMENT: The switches were inoperable for more than 6 months.

208

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 101078

Date: March 18, 1975

Title: Diesel Generator Fire at Fitzpatrick

The failure sequence was:

1. The operator noticed frequency oscillations on diesel generators B & D while diesels A & C were inoperable.
2. The operator dispatched to investigate the problem reported load noises and advised the control room operator to trip diesel D. Further investigation revealed the diesel was on fire and severely damaged.

Corrective action:

The reactor was placed in cold shutdown.

Design purpose of failed system or component:

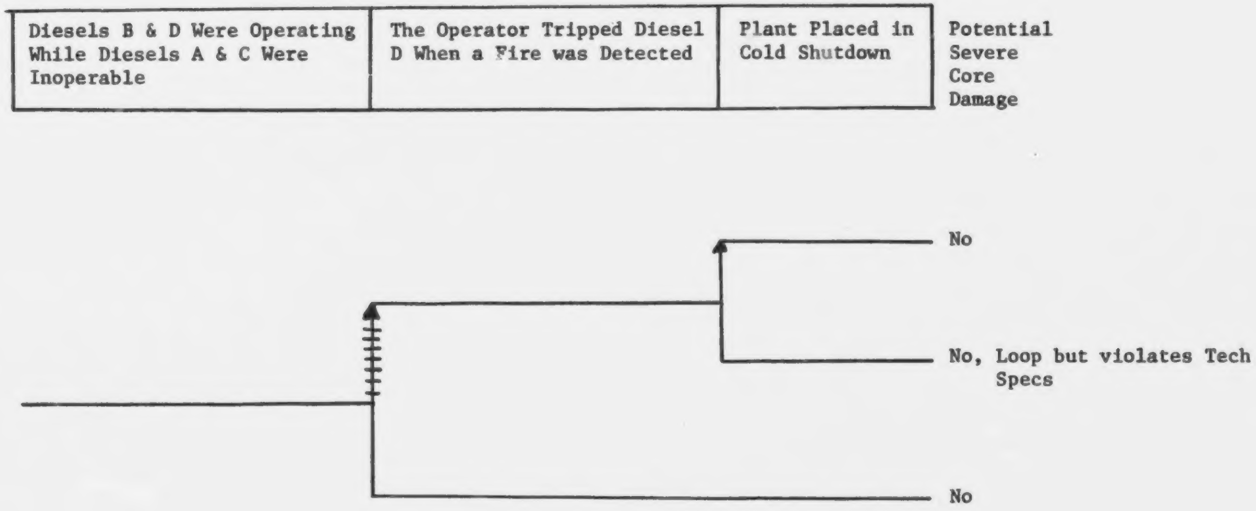
Emergency electric power provides a.c. power when offsite sources are unavailable.

Unavailability of system per WASH 1400:* Emergency Power: 1×10^{-3}/D

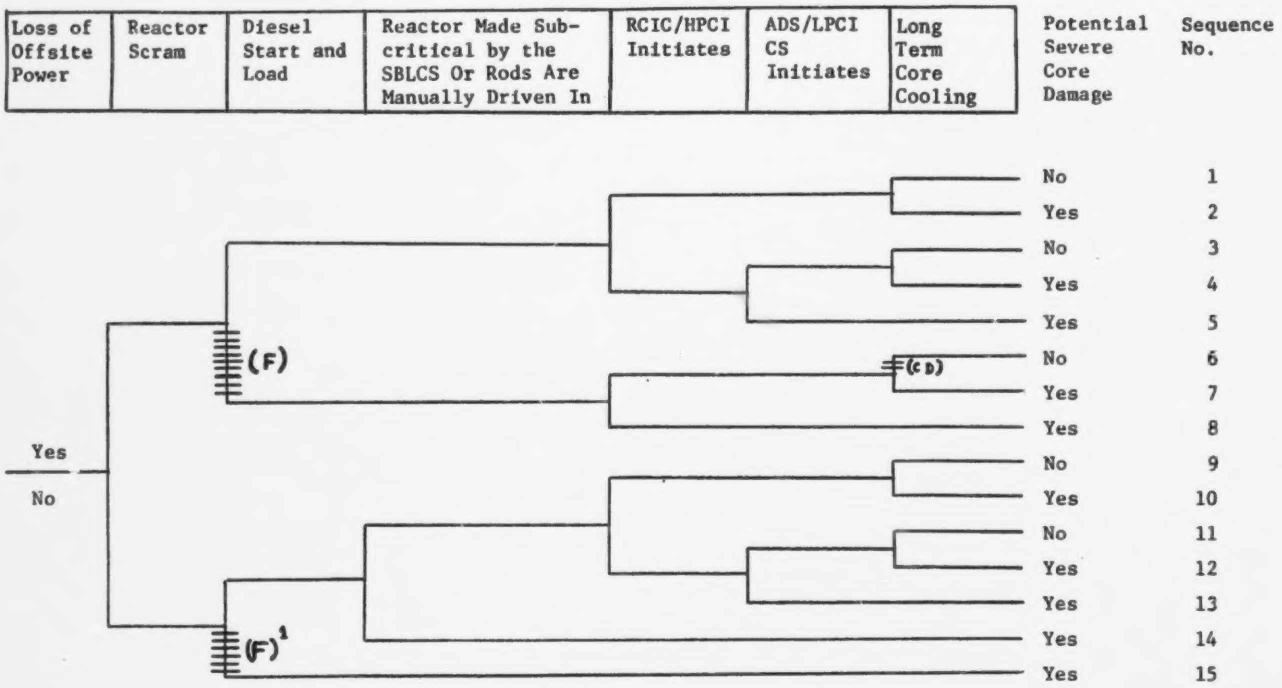
Unavailability of component per WASH 1400:* Diesel Generator failure to run: 3×10^{-4}/hr

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Diesels B & D Were Operating While Diesels A & C Were Inoperable
The Operator Tripped Diesel D When a Fire was Detected
Plant Placed in Cold Shutdown
Potential Severe Core Damage
No
No, Loop but violates Tech Specs
No

NSIC 101078 — Actual Occurrence of a Diesel Generator Fire at Fitzpatrick


Loss of Offsite Power
Reactor Scram
Diesel Start and Load
Reactor Made Subcritical by the SBLCS Or Rods Are Manually Driven In
RCIC/HPCI Initiates
ADS/LPCI CS Initiates
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
(F)
(F)1
(CD)
No 1
Yes 2
No 3
Yes 4
Yes 5
No 6
Yes 7
Yes 8
No 9
Yes 10
No 11
Yes 12
Yes 13
Yes 14
Yes 15

NSIC 101078 — Sequence of Interest for a Diesel Generator Fire at Fitzpatrick

[1] Since diesel generator B was operable the operator could have had either the control rod drive pumps or the standby liquid control system.

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 101078

DATE OF LER: March 26, 1975

DATE OF EVENT: March 18, 1975

SYSTEM INVOLVED: Emergency Power

COMPONENT INVOLVED: Diesel Generators

CAUSE: Diesels A & C were inoperable for unspecified reasons, D burned up.

SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE: Diesel Generator Fire at Fitzpatrick

REACTOR NAME: Fitzpatrick

DOCKET NUMBER: 50-333

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 821 MWe

REACTOR AGE: 0.3 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: Stone and Webster

OPERATORS: Power Authority of the State of New York

LOCATION: 8 miles NE of Oswego

DURATION: 12 (a) hours

PLANT OPERATING CONDITION: steady state power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________________

DISCOVERY METHOD: Failure during operation (not a loop)

COMMENT:

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 101444

Date: March 22, 1975

Title: Cable Tray Fire Causes Extensive Damage at Browns Ferry

The failure sequence was:

1. A fire started in cable spreading room (at approximately 12:20). About 20 minutes later anomalous behavior of the controls and instruments, especially in items associated with ECCS, were observed by the operator.
2. The operator scrammed the reactor at 12:51 pm.
3. Electrical boards supplying power to systems used to cool the reactor were lost, (12:54).
4. The MSIV automatically closed. This resulted in the following (1:00 pm):
   (a) The steam generated by the decay heat could not be dumped to the condenser.
   (b) The steam supply to the feedwater pump turbine was lost thus eliminating the remaining source of high pressure coolant injection.

(See attached page)

Corrective action:

1. The fire was extinguished.
2. The cables were repaired (1½ years).
3. Extensive changes were made to the cable spreading room, electrical penetrations, and insulating materials.

Design purpose of failed system or component:

Electric cables carry electric power and control signals to the various loads throughout the plant.

Unavailability of system per WASH 1400:* –

Unavailability of component per WASH 1400:* –

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

[redacted]

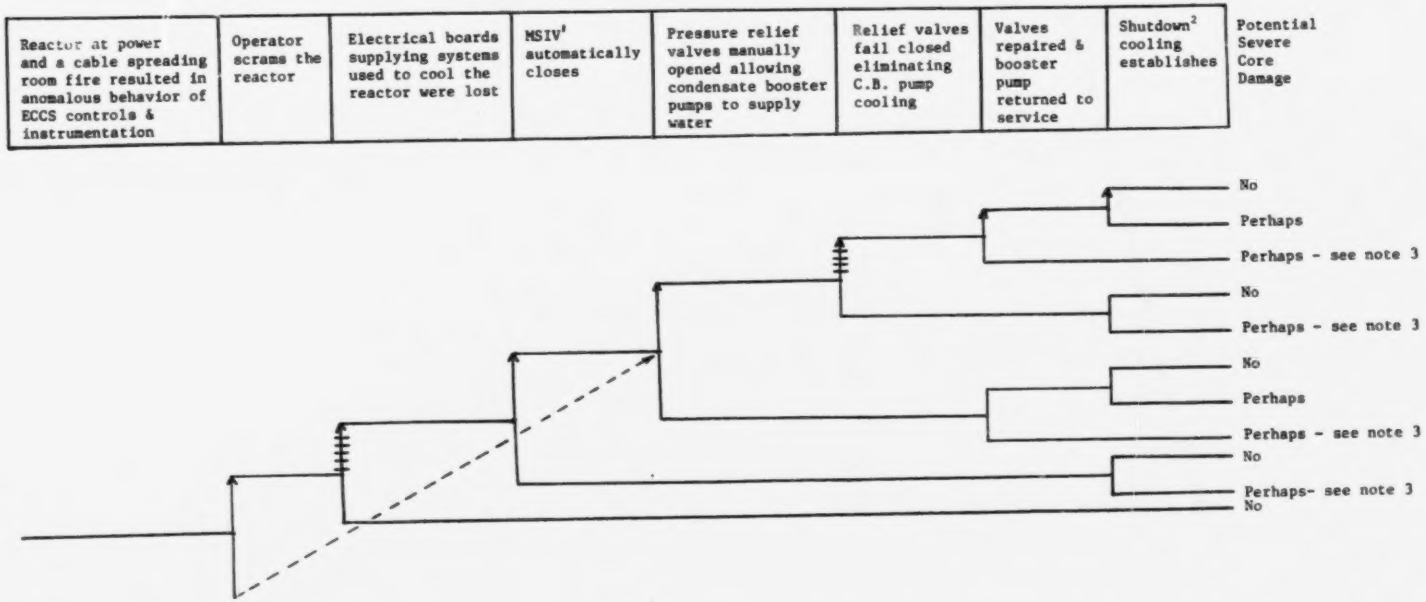
The failure sequence was: (continued)

5. Operable pressure relief valves were manually opened to depressurize the reactor to the point where the condensate booster pumps could be used to supply water, (1:20).

6. These valves later failed closed (6:00 pm), then the condensate booster pumps could no longer supply water. Only the control rod drive pumps were left operable.

7. The control of the relief valves was restored (9:50 pm) and the condensate booster pumps again supplied water.

8. Normal shutdown cooling was established at 9:10 am the next morning.


Reactor at power and a cable spreading room fire resulted in anomalous behavior of ECCS controls & instrumentation
Operator scrams the reactor
Electrical boards supplying systems used to cool the reactor were lost
MSIV¹ automatically closes
Pressure relief valves manually opened allowing condensate booster pumps to supply water
Relief valves fail closed eliminating C.B. pump cooling
Valves repaired & booster pump returned to service
Shutdown² cooling establishes
Potential Severe Core Damage
No
Perhaps
Perhaps - see note 3
No
Perhaps - see note 3
No
Perhaps
Perhaps - see note 3
No
Perhaps- see note 3
No

NSIC 101444 - Actual Occurrence of Cable Tray Fire at Browns Ferry 1

Notes

1. Closure of the MSIV had two immediate consequences.

   (a) The steam generated by decay heat could not be passed to the main condenser, but instead remained in the reactor.

   (b) The steam supply to the feedwater pump turbine was lost thus eliminating the remaining source of high pressure cooling water.

   The MSIV is designed to fail closed.

2. The MSIV closed shortly after 1:00 pm. Shutdown cooling wasn't restored until 4:30 am the next morning. Without adequate cooling over this period of time the core would have been severely damaged.

3. See the attached sheet. This information was obtained from *Nuclear Safety*, Vol. 17, No. 5, September-October 1976, p. 607.

Note 3

AVAILABILITY OF COOLING WATER

Following a reactor shutdown, radioactive material still present in the reactor fuel continues to generate a significant amount of heat, called decay heat, that must be removed to prevent fuel damage. There are several methods of removing this decay heat (Figs. 11 and 12).

1. Steam can be passed to the main condensers and water returned to the reactor to keep the fuel covered at all times.

2. The main-steam-line isolation valves can be closed to allow the reactor temperature and pressure to increase, thus causing relief valves to open and close automatically to maintain a relatively constant pressure. The steam discharge from the relief valves passes through pipes to a large pool of water called a suppression pool. For this type of operation, the reactor water level is automatically maintained above the fuel by means of high-pressure water makeup systems.

3. Relief valves can be opened by remote control to discharge steam to the suppression pool in order to reduce reactor pressure to a low value, thus permitting the use of low-pressure water makeup systems to maintain the water level above the fuel.

4. Decay heat can also be removed when the reactor is at low pressure by pumping the water directly through the residual-heat-removal heat exchangers.

After the reactors were shut down, supplying cooling-water makeup on Unit 1 was complicated because the fire in the electrical cables had caused a number of pieces of equipment to lose some or all of their capabilities. There was adequate high-pressure water makeup available on Unit 2 at all times to keep the fuel covered. The means of heat removal throughout the course of the incident was not critical because relief valves to transfer the decay heat to the suppression pool were available.

The fire damaged the control arrangements for the main-steam-line isolation valves in Unit 1, and the valves closed and could not be reopened. The decay heat was removed for a time using automatic operation of the relief valves with the reactor remaining at high pressure. However, the fire had also affected the two primary high-pressure water makeup systems provided for maintaining water level in an emergency. Therefore the operator chose to depressurize the reactor by remote control of the relief valves and to use the low-pressure water makeup systems that were still available.

It should be pointed out that, with the reactor at high pressure, there were other alternatives for obtaining makeup water to the reactor. A few examples of other alternatives are listed below:

1. The Unit 2 control-rod-drive (CRD) pump and a shared spare CRD pump could have been used in addition to the CRD pump on Unit 1.

2. The standby liquid-control pumps could have been made available by performing a manual valve alignment, actuating two valves, and manually restoring power to the pumps.

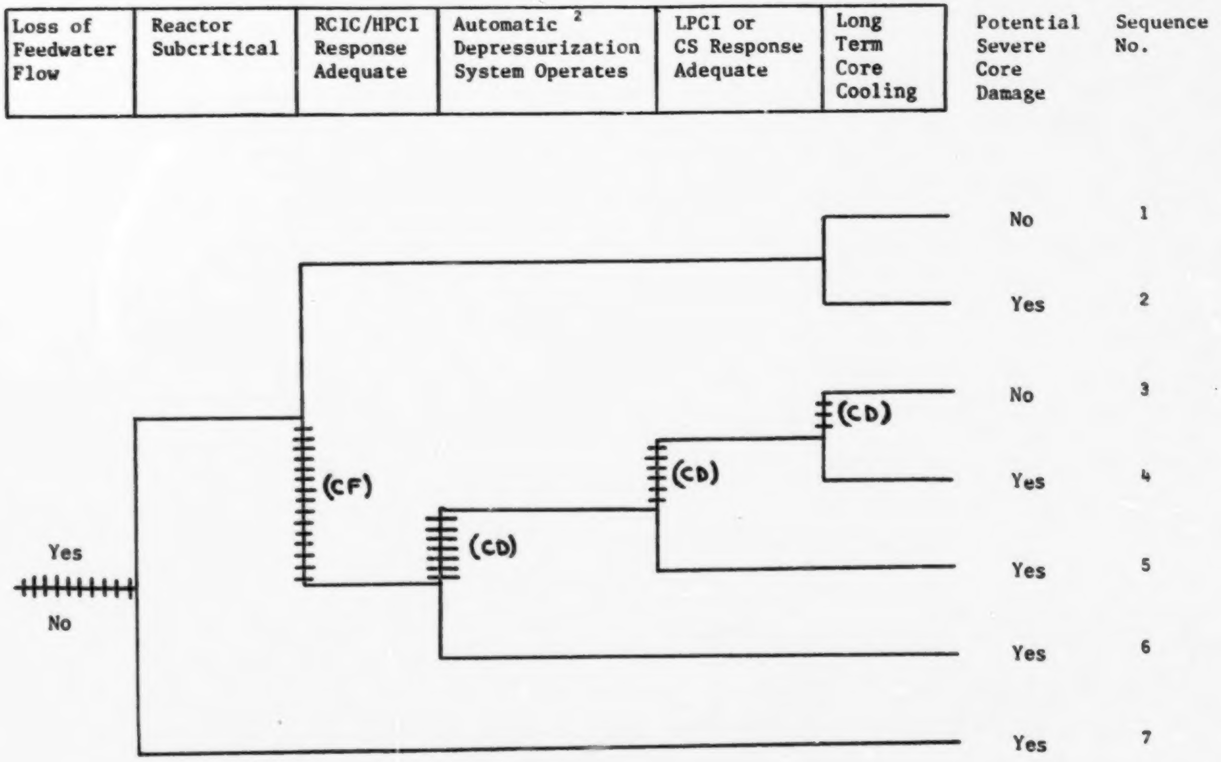
3. The reactor core-isolation cooling system could have been made available by installing a special short piece of pipe that was stored nearby.

During the period of depressurization, the water level in the core dropped but never fell below 1.22 m above the fuel. (Normal level is 5.08 m; the 1.22-m level is still 0.76 m above the level at which additional emergency cooling systems are actuated.) Once the reactor pressure was reduced below 2.4 MPa, a condensate booster pump and a condensate pump provided an adequate source of makeup water, and the normal water level was obtained. However, when stable low-pressure operation was attained, the operability of the relief valves being used to maintain low pressure was lost as a result of the loss of control air. The relief valves closed, pressure increased, and the availability of the low-pressure water makeup systems was lost. After about 3½ hr, operability of the relief valves was reestablished and low-pressure operation restored.

With low-pressure operation now secured, adequate makeup water could be supplied by one of the condensate pumps. In addition, two additional condensate booster pumps and two additional condensate pumps were available to the operator.

Another alternative would have been to use a nonstandard system configuration and manual valve alignment. Two residual-heat-removal pumps in Unit 2 could have been aligned to the Unit 1 reactor through a crosstie pipe, and, as an additional backup, river water could have been used from either of two available service-water pumps.

The point is that adequate cooling-water makeup was provided throughout the incident, and additional alternatives could have been used to provide makeup water with the reactor at either high or low pressure.


Loss of Feedwater Flow
Reactor Subcritical
RCIC/HPCI Response Adequate
Automatic 2 Depressurization System Operates
LPCI or CS Response Adequate
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
(CF)
(CD)
(CD)
(CD)
No 1
Yes 2
No 3
Yes 4
Yes 5
Yes 6
Yes 7

NSIC 101444 — Sequence of Interest of Cable Tray Fire at Browns Ferry 1

[1] Nonstandard techniques could have been used to make RCIC operable.

[2] The depressurization was manually initiated.

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: **101444**

DATE OF LER: **April 1, 1975**

DATE OF EVENT: **March 22, 1975**

SYSTEM INVOLVED: **electrical distribution**

COMPONENT INVOLVED: **electrical cable**

CAUSE: **fire, human error**

SEQUENCE OF INTEREST: **loss of feedwater**

ACTUAL OCCURRENCE: **cable tray fire causes extensive damage**

REACTOR NAME: **Browns Ferry 1**

DOCKET NUMBER: **50-259**

REACTOR TYPE: **BWR**

DESIGN ELECTRICAL RATING: **1065** MWe

REACTOR AGE: **1.59** yr

VENDOR: **GE**

ARCHITECT-ENGINEERS: **TVA**

OPERATORS: **TVA**

LOCATION: **10 miles NW of Decatur, Alabama**

DURATION: **N/A**

PLANT OPERATING CONDITION: **100% power**

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________________

DISCOVERY METHOD: **operational event**

COMMENT: —

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: **102146**

Date: **April 9, 1975**

Title: **Loss of Offsite Power at Brunswick 2**

The failure sequence was:

1. **The plants 1A and 2A 230kv buses had been incorrectly tied together.**
2. **With the reactor in cold shutdown, a single phase to ground fault caused the lockout relays for busses 1A and 2A to trip.**
3. **The diesels started and loaded as expected.**
4. **The relays were reset and offsite power restored (∿4 min).**

Corrective action:

This problem was corrected.

Design purpose of failed system or component:

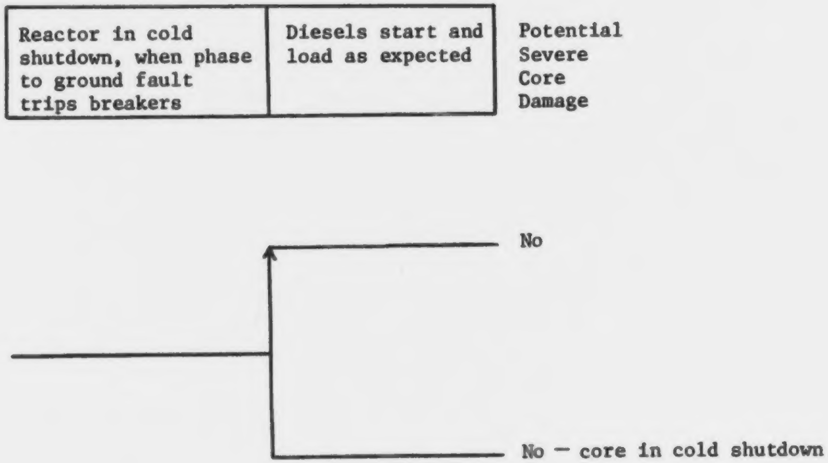
The offsite power source provides power to the plant when the unit generator isn't functioning.

Unavailability of system per WASH 1400:*

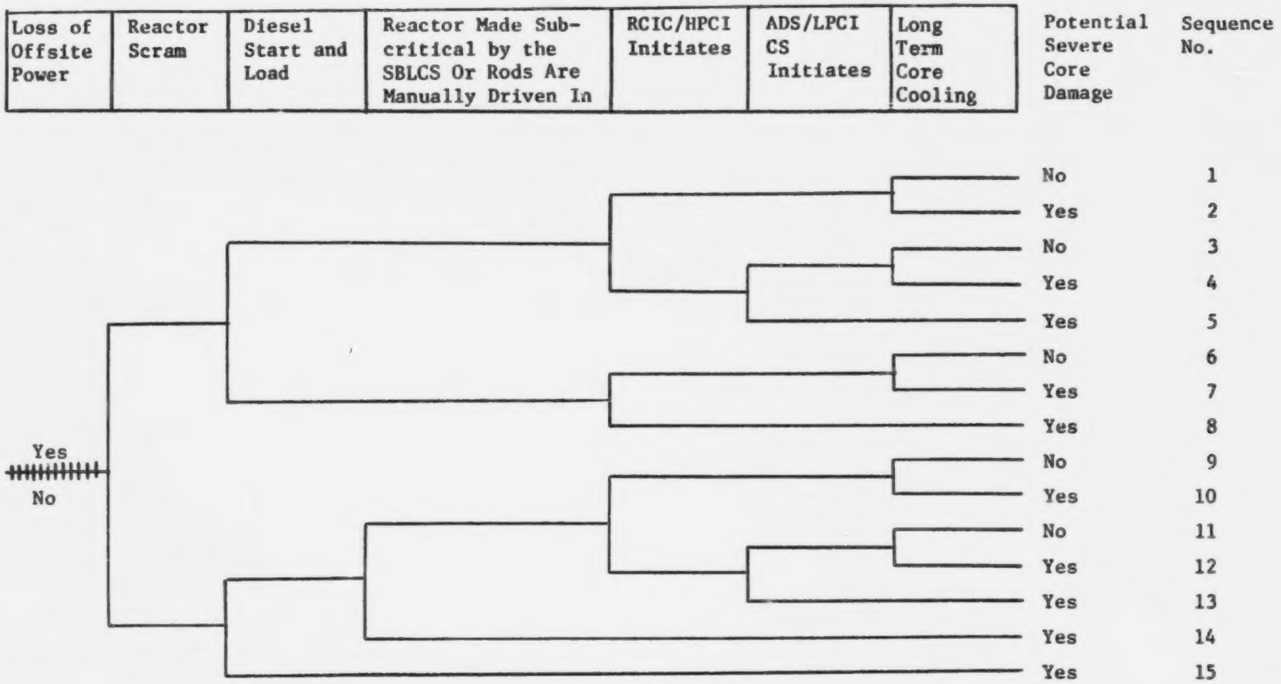
Unavailability of component per WASH 1400:* **maintenance error: 3×10^{-3}/D**

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor in cold shutdown, when phase to ground fault trips breakers
Diesels start and load as expected
Potential Severe Core Damage
No
No — core in cold shutdown

NSIC 102146 — Actual Occurrence of Loss of Offsite Power at Brunswick 2


Loss of Offsite Power
Reactor Scram
Diesel Start and Load
Reactor Made Subcritical by the SBLCS Or Rods Are Manually Driven In
RCIC/HPCI Initiates
ADS/LPCI CS Initiates
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
No 1
Yes 2
No 3
Yes 4
Yes 5
No 6
Yes 7
Yes 8
No 9
Yes 10
No 11
Yes 12
Yes 13
Yes 14
Yes 15

NSIC 102146 — Sequence of Interest for Loss of Offsite Power at Brunswick 2

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 102146

DATE OF LER: April 9, 1975

DATE OF EVENT: March 26, 1975

SYSTEM INVOLVED: Electric power

COMPONENT INVOLVED: relays

CAUSE: improper tying together of the 1A and 2A 230kv buses, human error

SEQUENCE OF INTEREST: loss of offsite power

ACTUAL OCCURRENCE: loss of offsite power

REACTOR NAME: Brunswick 2

DOCKET NUMBER: 50-324

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 821 MWe

REACTOR AGE: .05 yr

VENDOR: GE

ARCHITECT-ENGINEERS: United Engineers and Constructors

OPERATORS: Carolina Power and Light

LOCATION: 3 miles N of South Port, NC

DURATION: N/A

PLANT OPERATING CONDITION: cold shutdown

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________

DISCOVERY METHOD: operation

COMMENT: –

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 103002

Date: April 29, 1975

Title: Multiple Valve Failures and RCIC Inoperable at Brunswick 2

The failure sequence was:

1. The reactor was operating at 10% power and RCIC was inoperable.
2. The "B" safety valve stuck open. (reason unspecified)
3. The operator failed to scram the reactor as specified by emergency procedures.
4. HPCI manually initiated but was shutdown due to high level in the torus. (This was done to prevent torus water from entering the core following expected HPCI suction switchover to torus on high torus level.
5. Reactor autoscram as MSIV's were manually closed to conserve reactor water.
6. An attempt was made to place the "B" RHR loop in service to cool the torus during the relief valve blowdown, however this failed. The service water supply valve to the RHR heat exchanger failed to open.

Corrective action:

All relief valves except the "A" relief valves were removed, overhauled, and reinstalled. The "A" valve had just been overhauled and was operating correctly.

Design purpose of failed system or component:

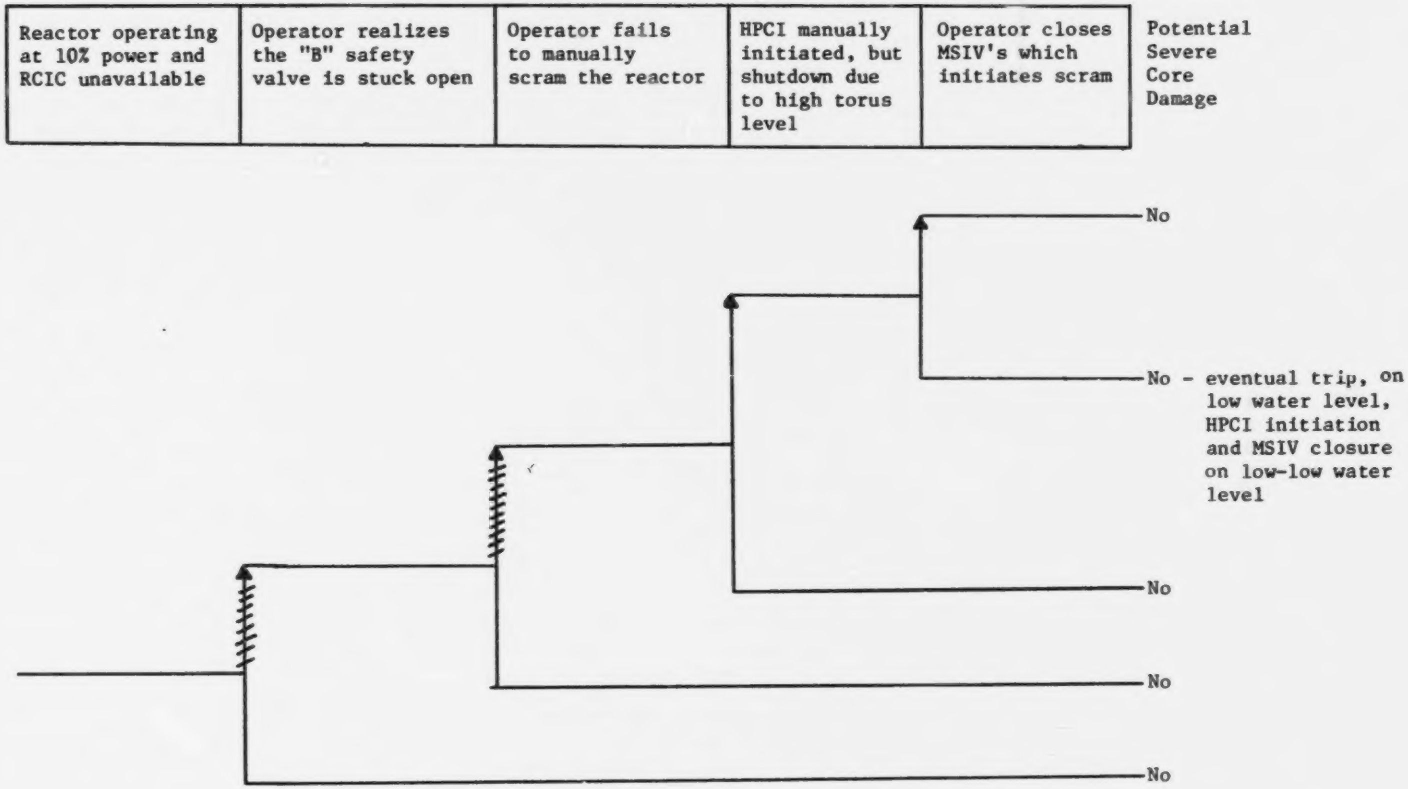
1. RCIC provides for core cooling when feedwater is unavailable.
2. RHR provides for long term core cooling.
3. The relief valves provide for a controlled depressurization of the reactor.

Unavailability of system per WASH 1400:* RCIC: 8×10^{-2}/D

Unavailability of component per WASH 1400:* RHR valve (SOV): 1×10^{-3}/D
relief valve, failure to close: 1×10^{-2}/D

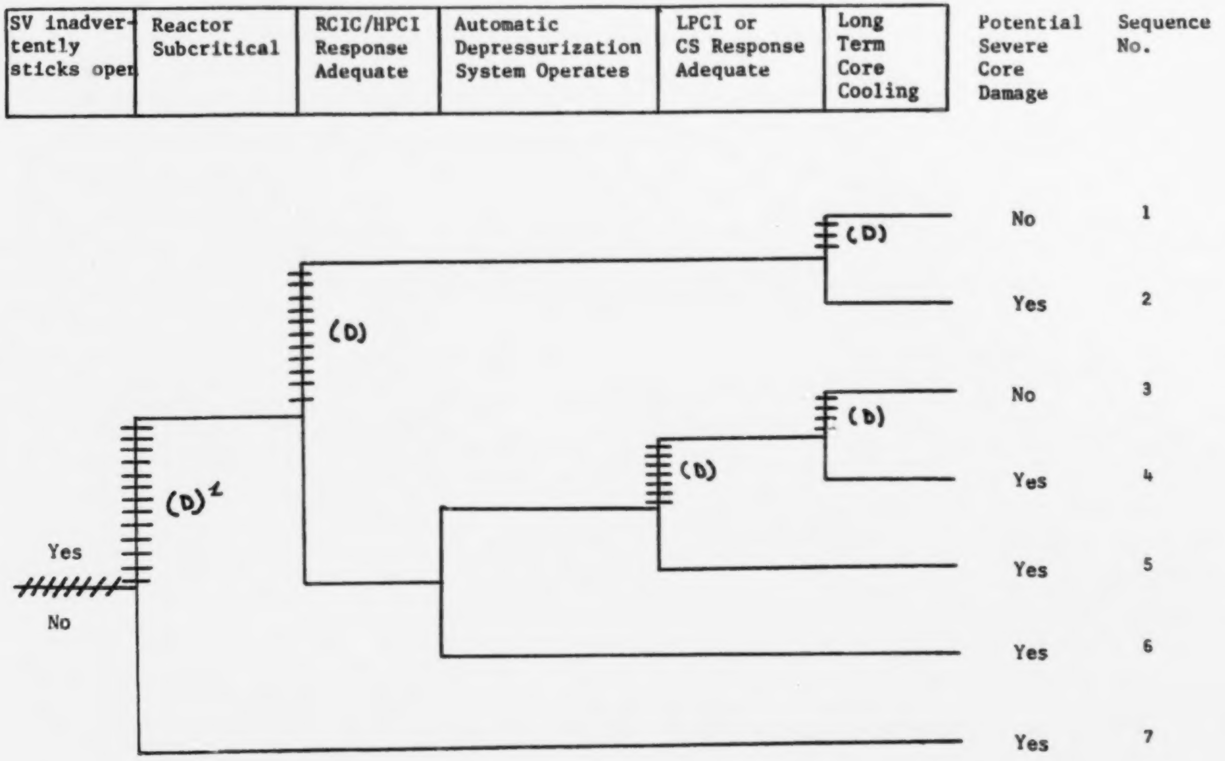
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor operating at 10% power and RCIC unavailable
Operator realizes the "B" safety valve is stuck open
Operator fails to manually scram the reactor
HPCI manually initiated, but shutdown due to high torus level
Operator closes MSIV's which initiates scram
Potential Severe Core Damage
No
No - eventual trip, on low water level, HPCI initiation and MSIV closure on low-low water level
No
No
No

NSIC 103002 — Actual Occurrence of Multiple Valve Failures and RCIC Inoperable at Brunswick 2

BEST DOCUMENT AVAILABLE


SV inadvertently sticks open
Reactor Subcritical
RCIC/HPCI Response Adequate
Automatic Depressurization System Operates
LPCI or CS Response Adequate
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
(D)1
(D)
(D)
(D)
(D)
No 1
Yes 2
No 3
Yes 4
Yes 5
Yes 6
Yes 7

NSIC 103002 — Sequence of Interest for Multiple Valve Failures and RCIC Inoperable at Brunswick 2

[1] The operator failed to manually scram the reactor as emergency instruction requires.

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 103002

DATE OF LER: May 16, 1975

DATE OF EVENT: April 29, 1975

SYSTEM INVOLVED: RCIC, RHR, Reactor coolant system (RCS)

COMPONENT INVOLVED: isolation & relief valves

CAUSE: mechanical failure of the valve pilot, operator error

SEQUENCE OF INTEREST: loss of feedwater flow

ACTUAL OCCURRENCE: multiple valve failures and RCIC inoperable at Brunswick 2

REACTOR NAME: Brunswick 2

DOCKET NUMBER: 50-324

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 821 MWe

REACTOR AGE: .1 yr

VENDOR: GE

ARCHITECT-ENGINEERS: United Engineers & Constructors

OPERATORS: Carolina Power & Light

LOCATION: 3 miles N of Southport, NC

DURATION: N/A

PLANT OPERATING CONDITION: 10% power

SAFETY FEATURE TYPE OF FAILURE: [✓ (a)] inadequate performance; (b) failed to start; (c) made inoperable; (d) ________________________

DISCOVERY METHOD: Operational event

COMMENT: –

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 103077

Date: May 12, 1975

Title: Reactor Coolant Pump Seal System Fails at . binson 2

The failure sequence was:

1. The plant was at power and diluting for xenon control. The number 1 seal for the "C" RCP was exhibiting gradual flow variations with all additions of RCS makeup water.
2. The "C" RCP No. 1 seal leakoff spiked several times, oscillated full range several times, then stabilized with a seal flow greater than 6 gpm.
3. Plant load was reduced at 10%/minute to 36% power and the "C" pump idled.
4. Reactor trip occurred due to turbine trip on high steam generator level, resulting from the rapid load reduction and use of steam dump for cooldown.

(See additional page)

Corrective action;

1. New seals were installed on RCP "C".

Design purpose of failed system or component:

1. The RCP seals provide a barrier at the point at which the pump shaft enters the RCS which prevents reactor coolant from escaping into the containment.

Unavailability of system per WASH 1400:

Unavailability of component per WASH 1400:[*] not considered in WASH-1400.

---

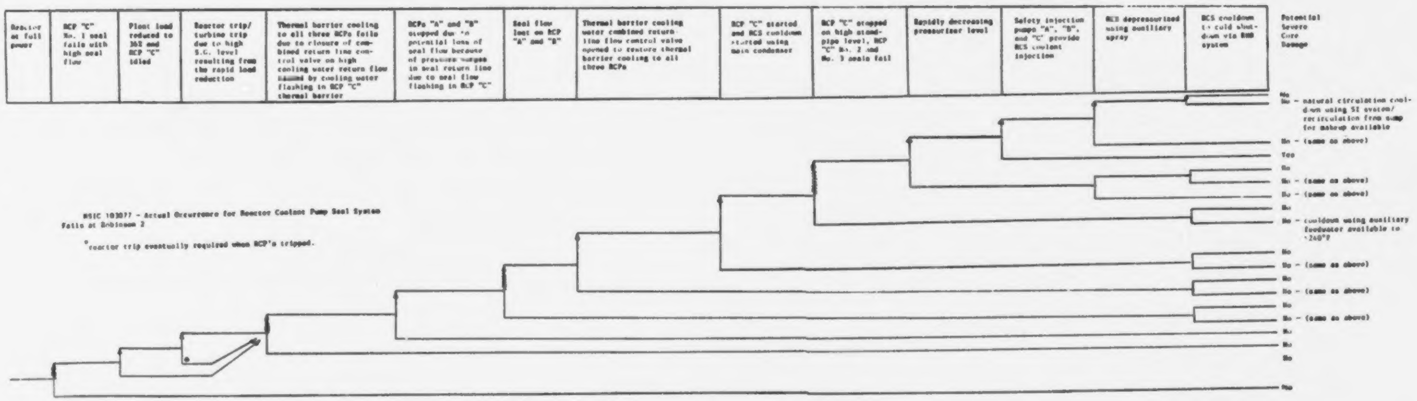
[*] Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

[redacted]

The failure sequence was: (Continued)

5. The flow control valve in the combined return line from the three RCP thermal barrier cooling lines closed due to high flow caused by cooling water flashing in the "C" RCP thermal barrier. The flashing was caused by hot primary coolant flowing upward through the "C" RCP thermal barrier. Closure of the flow control valve resulted in loss of thermal barrier cooling in all three RCPs.
6. Reactor coolant pumps "A" and "B" were stopped because flashing in the seal return line threatened to cause loss of seal flow due to pressure surges. The flashing was caused by the high primary flow rate through the No. 1 seal of RCP "C".
7. The RCP "C" No. 1 seal return flow isolation valve was closed to decrease pressure surges in the letdown line.
8. Seal flow was lost on RCP "A" and "B".
9. Leakage through RCP "C" No. 2 seal resulted in high Reactor Cooldown Drain Tank (RCDT) pressures. The RCDT was drained to the containment sump.
10. The flow control valve in the combined return line from the three RCP thermal barriers was blocked open, restoring thermal barrier cooling on all three RCPs.
11. Reactor coolant pump "C" was started with increased seal flow and RCS cooldown was started using condenser dump.
12. A high standpipe alarm was received for RCP "C" and the pump was stopped.
13. Rapidly falling pressurizer level indicated failure of RCP "C" No. 2 and No. 3 seals.
14. Safety injection pumps "A", "B", and "C" were started to makeup for rapidly decreasing pressurizer level. Pressurizer level stabilized and SI pump "C" was stopped.
15. Auxiliary pressurizer spray was used to reduce plant pressure to the operating pressure of the RHR system. During this pressure reduction, the SI Accumulators partially discharged into the RCS before their isolation valves were closed.
16. Cooldown via the RHR system was used to achieve cold shutdown. Final conditions were: RCS pressure — 0 psig; RCS temperature — <200°F; Containment temperature — 100°F; 132,500 gallons of water in containment.

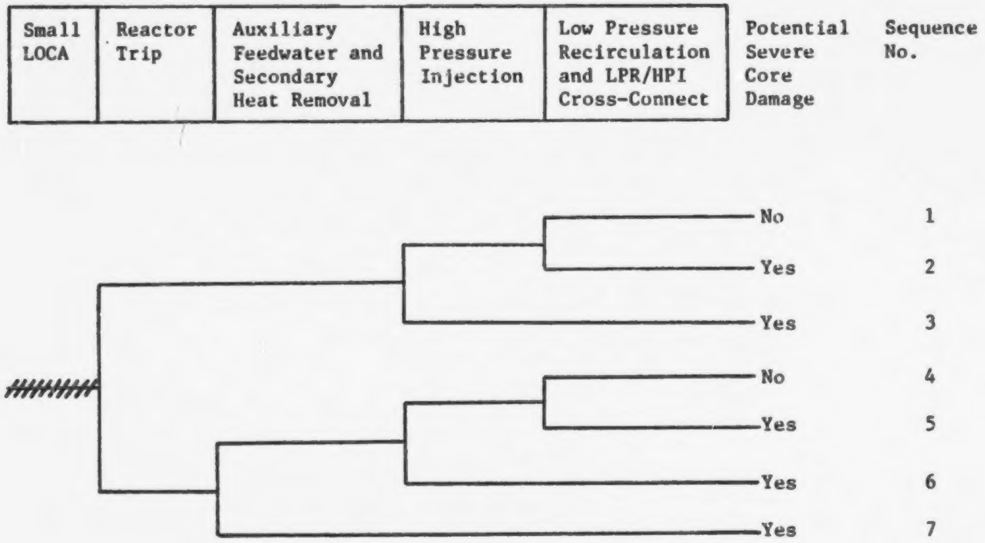
NOTE: Based on system response to the use of auxiliary spray, CP&L concluded that a second steam bubble existed in the system, probably in the steam generator tubes, since little gas or steam escaped when the vessel head was later vented.



NSIC 103077 - Actual Occurrence for Reactor Coolant Pump Seal System Fails at Robinson 2

*reactor trip eventually required when RCP's tripped.

Reactor at full power
RCP "C" No. 1 seal fails with high seal flow
Plant load reduced to 36% and RCP "C" idled
Reactor trip/ turbine trip due to high S.G. level resulting from the rapid load reduction
Thermal barrier cooling to all three RCPs fails due to closure of combined return line control valve on high cooling water return flow caused by cooling water flashing in RCP "C" thermal barrier
RCPs "A" and "B" stopped due to potential loss of seal flow because of pressure surges in seal return line due to seal flow flashing in RCP "C"
Seal flow lost on RCP "A" and "B"
Thermal barrier cooling water combined return line flow control valve opened to restore thermal barrier cooling to all three RCPs
RCP "C" started and RCS cooldown started using main condenser
RCP "C" stopped on high standpipe level, RCP "C" No. 2 and No. 3 seals fail
Rapidly decreasing pressurizer level
Safety injection pumps "A", "B", and "C" provide RCS coolant injection
RCS depressurized using auxiliary spray
RCS cooldown to cold shutdown via RHR system
Potential Severe Core Damage
No
No - natural circulation cooldown using SI system/ recirculation from sump for makeup available
No - (same as above)
Yes
No
No - (same as above)
No - (same as above)
No
No - cooldown using auxiliary feedwater available to ~240°F
No
No - (same as above)
No
No - (same as above)
No
No - (same as above)
No
No
No
No


Small LOCA
Reactor Trip
Auxiliary Feedwater and Secondary Heat Removal
High Pressure Injection
Low Pressure Recirculation and LPR/HPI Cross-Connect
Potential Severe Core Damage
Sequence No.
No 1
Yes 2
Yes 3
No 4
Yes 5
Yes 6
Yes 7

NSIC 103077 — Sequence of Interest for Reactor Coolant Pump Seal System Fails at Robinson 2

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 103077

DATE OF LER: May 12, 1975

DATE OF EVENT: May 1, 1975

SYSTEM INVOLVED: Reactor Coolant System

COMPONENT INVOLVED: RCP seals

CAUSE: Seal failure due to a defective seal

SEQUENCE OF INTEREST: Small LOCA

ACTUAL OCCURRENCE: RCP Seal Failure

REACTOR NAME: H. B. Robinson 2

DOCKET NUMBER: 50-261

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 700 MWe

REACTOR AGE: 4.9 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Ebasco

OPERATORS: Carolina Power & Light Co.

LOCATION: **5 miles NW of Hartsville, SC**

DURATION: N/A

PLANT OPERATING CONDITION: At full power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; (c) made inoperable; [✓ (d) failed to function]

DISCOVERY METHOD: during operation

COMMENT: –

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 103111

Date: April 19, 1975

Title: Condenser Circulating Water Valves Fail to Open at Oconee 3

The failure sequence was:

1. Due to inadequate procedure, switches which select valve operation for the condenser discharge valves were improperly positioned, preventing discharge valve closure during loss-of-offsite power.
2. During a CCW system gravity and recirculation flow test, three discharge valves failed to reposition. Since the emergency valves are interlocked with the discharge valves, the emergency valves did not open when the discharge valves failed to close.

Corrective action;

1. The operating procedure was revised to specify the required position of the control switches.

Design purpose of failed system or component:

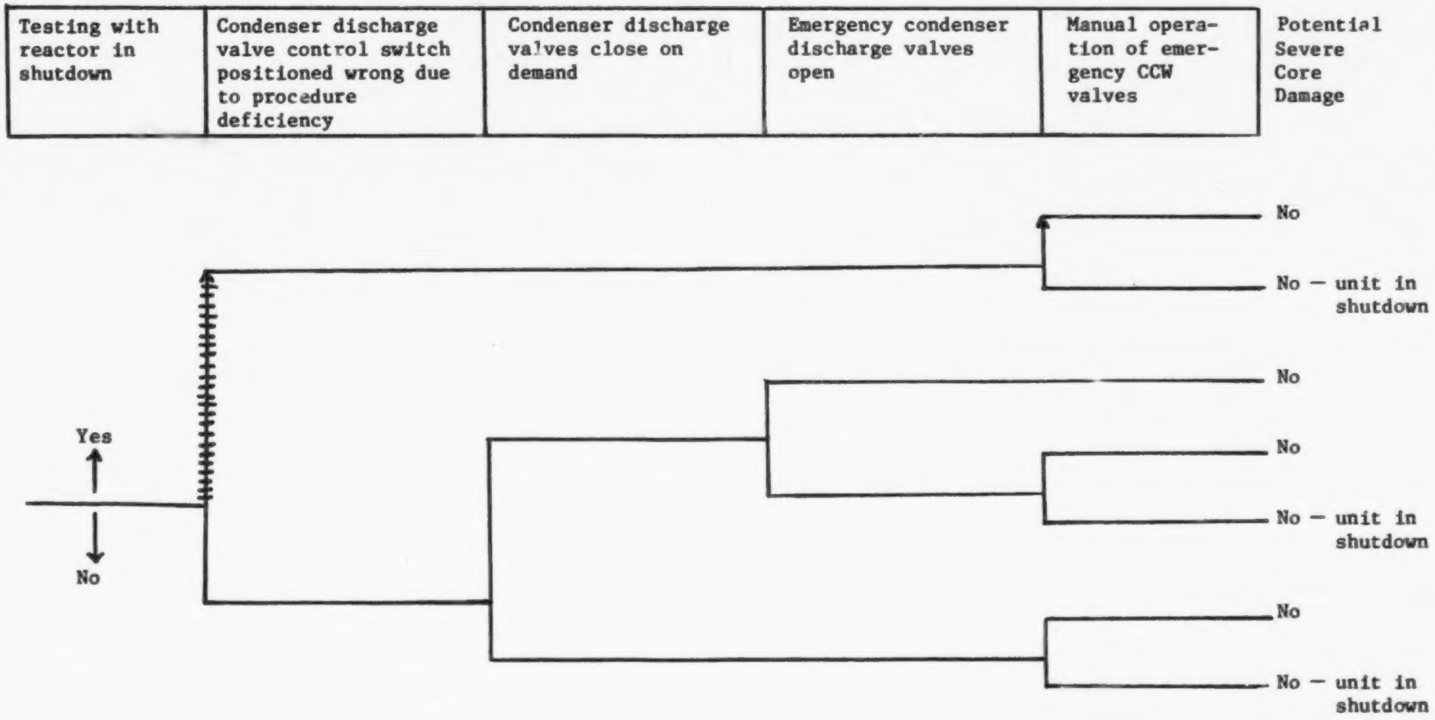
1. The emergency condenser cooling water system valves are provided to connect the condenser to siphon flow piping which provides an alternate cooling water flow in the event the normal circulating water system is not in operation.

Unavailability of system per WASH 1400:* system is unique to Oconee and hence not considered in WASH-1400.

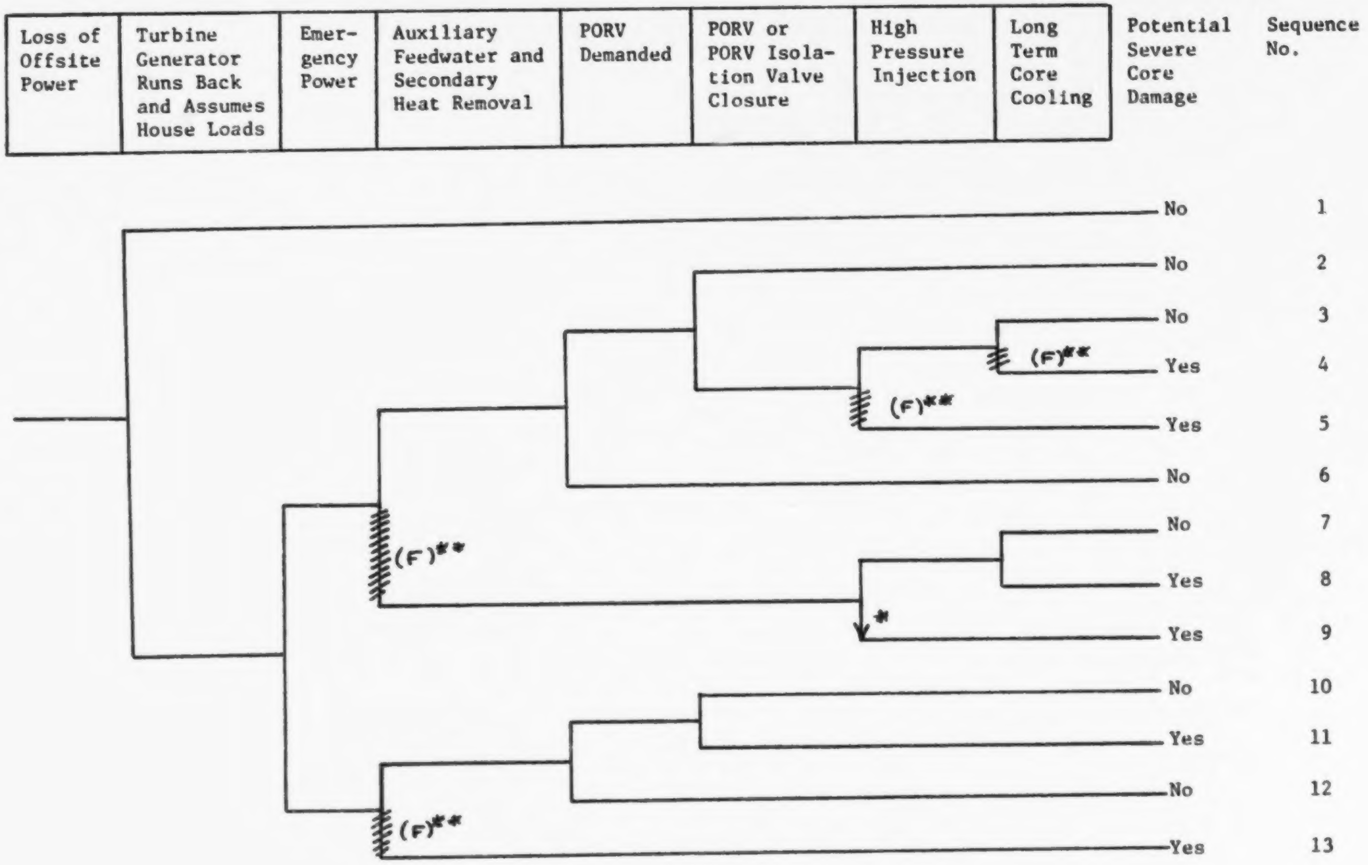
Unavailability of component per WASH 1400:* general human error of omission: 3×10^{-3}

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Testing with reactor in shutdown
Condenser discharge valve control switch positioned wrong due to procedure deficiency
Condenser discharge valves close on demand
Emergency condenser discharge valves open
Manual operation of emergency CCW valves
Potential Severe Core Damage
Yes
No
No
No — unit in shutdown
No
No
No — unit in shutdown
No
No — unit in shutdown

NSIC 103111 — Actual Occurrence for Condenser Circulating Water Valves Failure to Close at Oconee 3 (Emergency valves interlocked to open on closure of normal discharge valves.)


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
No 3
(F)**
Yes 4
(F)**
Yes 5
No 6
No 7
(F)**
Yes 8
*
Yes 9
No 10
Yes 11
No 12
(F)**
Yes 13

NSIC 103111 — Sequence of Interest for Condenser Circulating Water Valves Failure to Close at Oconee 3

*not included in mitigation procedures.

**success requires manual operation of emergency discharge valves to establish circulating water flow.

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 103111

DATE OF LER: May 30, 1975

DATE OF EVENT: April 19, 1975

SYSTEM INVOLVED: Condenser circulating water

COMPONENT INVOLVED: discharge valves

CAUSE: Procedures did not designate proper position of switches, operator error.

SEQUENCE OF INTEREST: Loss of Offsite Power

ACTUAL OCCURRENCE: During test, the condenser discharge valves failed to close & thus emergency discharge valves did not open.

REACTOR NAME: Oconee 3

DOCKET NUMBER: 50-287

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 887 MWe

REACTOR AGE: 1.6 yr

VENDOR: B&W

ARCHITECT-ENGINEERS: Duke Power Co.

OPERATORS: Duke Power Co.

LOCATION: 30 miles west of Greenville, SC

DURATION: 360(a) hours

PLANT OPERATING CONDITION: Shutdown

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________________

DISCOVERY METHOD: Test

COMMENT: A common mode failure of the six discharge valves, combined with operator failure to manually open the emergency cooling discharge valves will result in the loss of the ultimate heat sink.

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 103207

Date: June 4, 1975

Title: Diesel Generator Breaker and Bus Stripping Relay Malfunction at Turkey Point 4

The failure sequence was:

1. With Unit 4 in a refueling shutdown and Unit 3 in hot shutdown, an Engineered Safeguards and Emergency Power System Integrated Test was in progress.
2. The Unit 4 safety-related buses were manually deenergized and safety injection and containment isolation signals initiated.
3. Diesel generator 3 started, came up to rated speed and then shutdown due to closure of its lockout relay as a result of failure of output breaker 4A to close.
4. Diesel generator 4 started and came up to speed. However, its 4B breaker did not close due to loads not being stripped from the 4B bus because of a load stripping relay failure.

Corrective action;

1. Offsite power was restored and the necessary equipment restarted.
2. The 4A diesel generator breaker and the load stripping relay were replaced.

Design purpose of failed system or component:

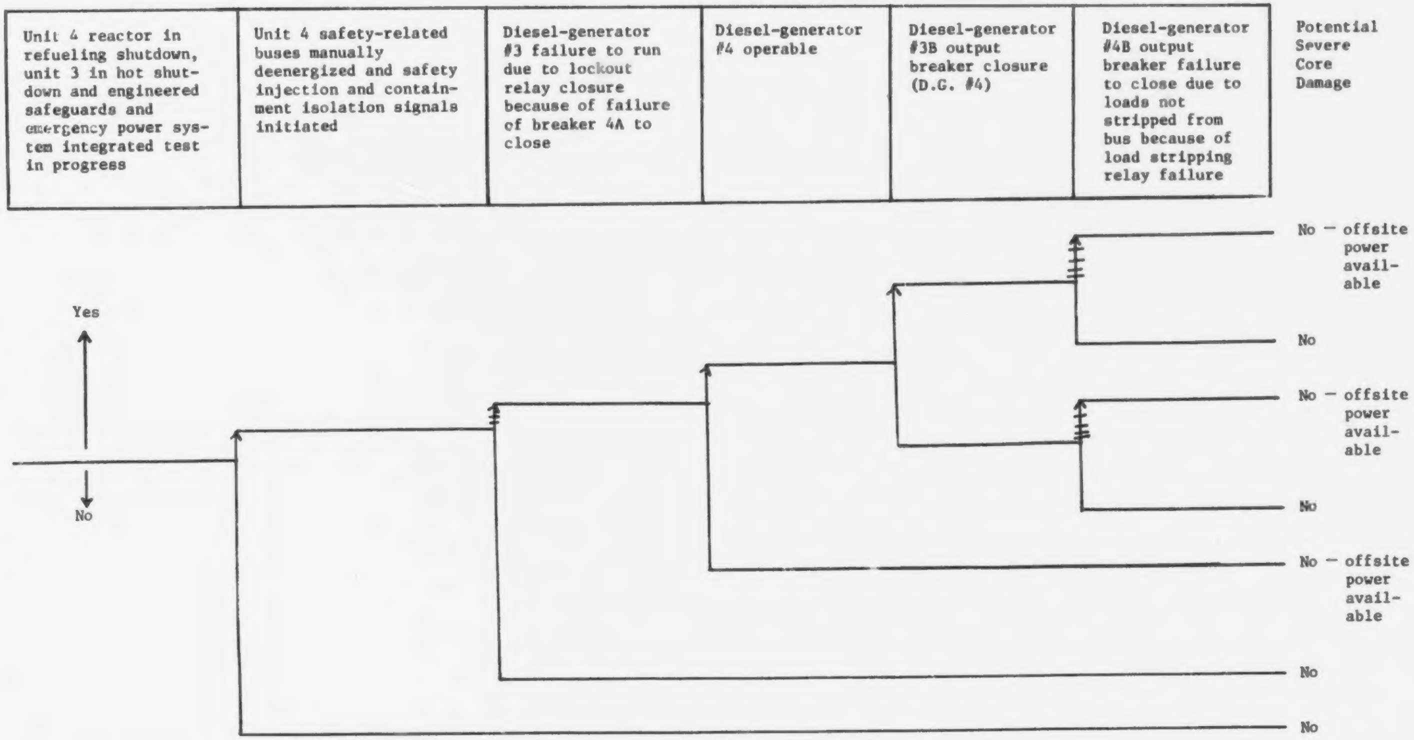
1. The diesel generator breaker loads the diesel generator output on a safety related bus.
2. The stripping relay removes all loads from a bus prior to the diesel generator being loaded on the bus.
3. The diesel generators provide emergency power to safety-related loads in the event normal power is unavailable.

Unavailability of system per WASH 1400:* emergency power: 1×10^{-2}/D

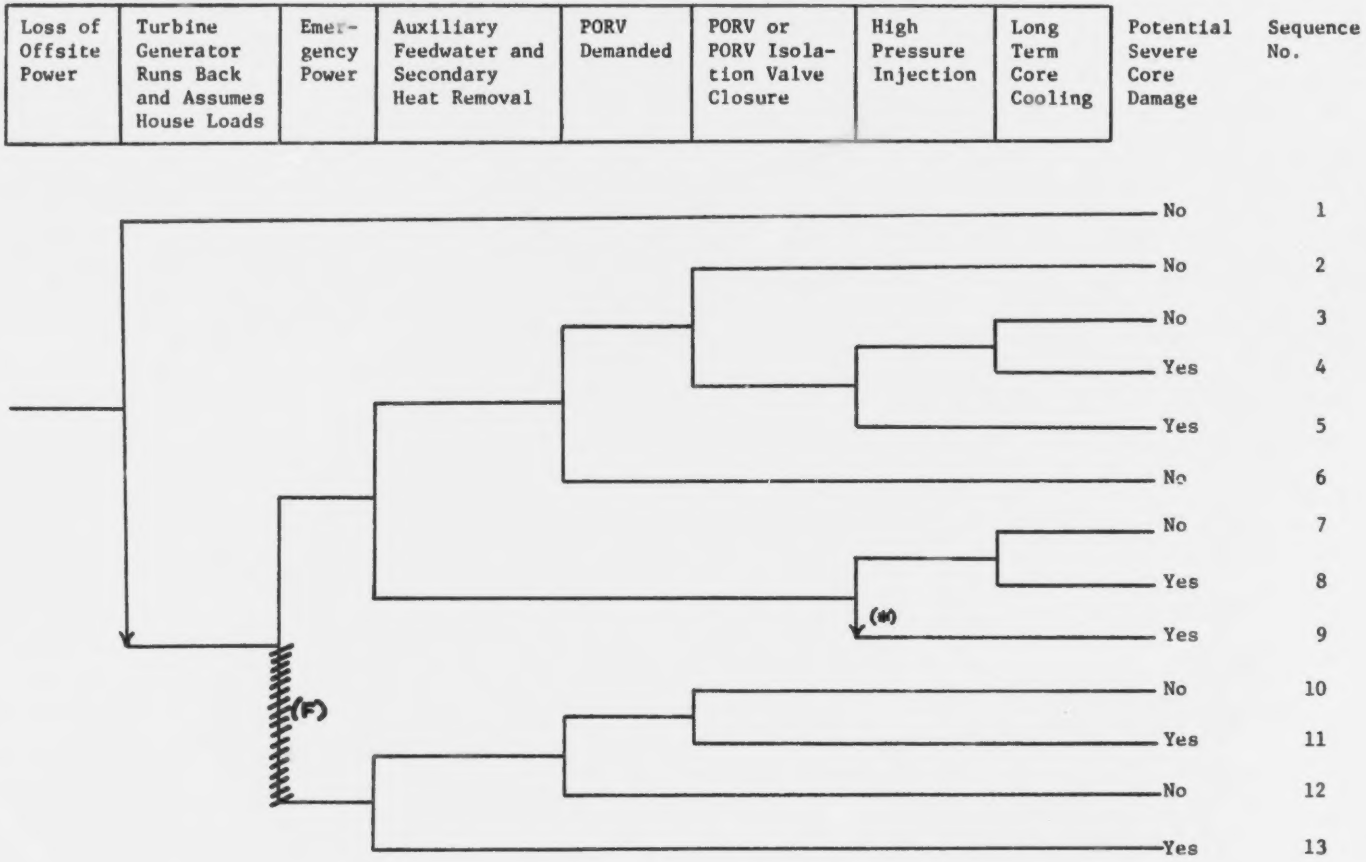
Unavailability of component per WASH 1400:* relay, failure to operate: 1×10^{-4}/D

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Unit 4 reactor in refueling shutdown, unit 3 in hot shutdown and engineered safeguards and emergency power system integrated test in progress
Unit 4 safety-related buses manually deenergized and safety injection and containment isolation signals initiated
Diesel-generator #3 failure to run due to lockout relay closure because of failure of breaker 4A to close
Diesel-generator #4 operable
Diesel-generator #3B output breaker closure (D.G. #4)
Diesel-generator #4B output breaker failure to close due to loads not stripped from bus because of load stripping relay failure
Potential Severe Core Damage
Yes
No
No — offsite power available
No
No — offsite power available
No
No — offsite power available
No
No

NSIC 103207 — Actual Occurrence for Diesel Generator Breaker and Bus Stripping Relay Malfunction at Turkey Point 4


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
Yes 9
No 10
Yes 11
No 12
Yes 13
(F)
(*)

NSIC 103207 — Sequence of Interest for Diesel Generator Breaker and Bus Stripping Relay Malfunction at Turkey Point 4

*use of HPI following AFW failure not included in mitigation procedures.

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 103207

DATE OF LER: June 4, 1975

DATE OF EVENT: May 25, 1975

SYSTEM INVOLVED: electric power

COMPONENT INVOLVED: diesel generator breaker, load stripping relay

CAUSE: The breaker failed to close, the relay failed to strip loads.

SEQUENCE OF INTEREST: Reactor trip with loss of offsite power

ACTUAL OCCURRENCE: Failure of breaker and relay to operate during engineered safeguards and emergency power system integrated test.

REACTOR NAME: Turkey Point 4

DOCKET NUMBER: 50-251

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 693 MWe

REACTOR AGE: 2.1 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Florida Power & Light Co.

LOCATION: 25 miles south of Miami, Fla.

DURATION: 360(a) hours

PLANT OPERATING CONDITION: Unit 3 in hot shutdown. Unit 4 in refueling shutdown.

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start;
(c) made inoperable; [✓ (d) failed to operate]

DISCOVERY METHOD: during testing

COMMENT: —

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 103892

Date: June 27, 1975

Title: Excessive Reactor Coolant Cooldown Rate at Oconee 3

The failure sequence was:

1. During a power reduction, an operator initiated a secondary side transient, which caused a primary side pressure/temperature transient and opened the pilot actuated pressurizer relief valve.
2. The relief valve failed to close (due to boric acid crystal buildup) and did not indicate open in the control room.
3. Reactor trip and high pressure injection occurred.
4. The operator closed the PORV block valve, re-opened it because of increasing pressurizer level, and finally closed it when RC pressure reached 800 psig, terminating the transient.
5. The relief valve quench tank rupture disk blew out, damaging some pressurizer insulation.

Corrective action:

1. Procedures revised to prevent the initiation of a secondary side transient.
2. The unit 1 and unit 2 PORVs were to be examined for boric acid crystal buildup.
3. The PORV will be test cycled prior to each startup.
4. Operating personnel were advised that closure of the block valve was the corrective action for this occurrence.

Design purpose of failed system or component:

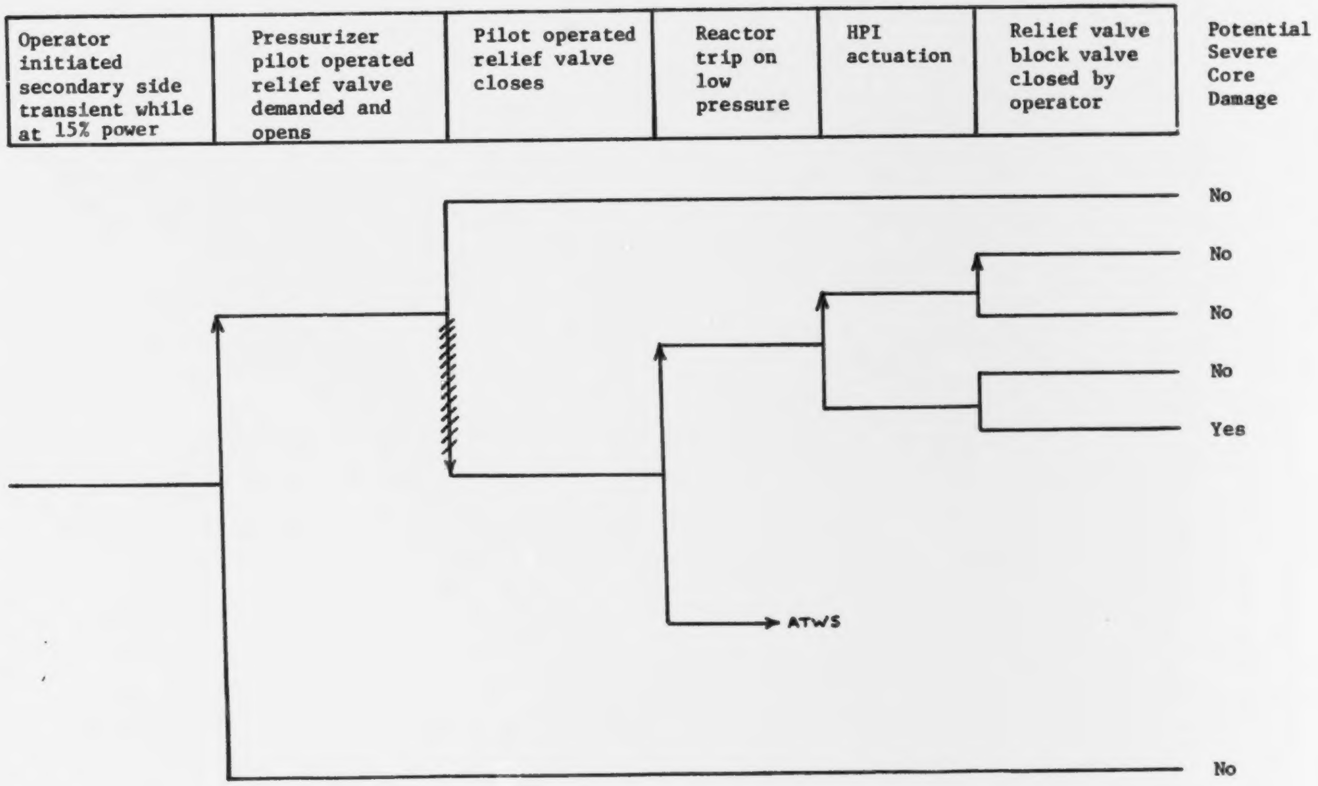
1. The valve provides pressure relief protection for the RCS.

Unavailability of system per WASH 1400:* N/A

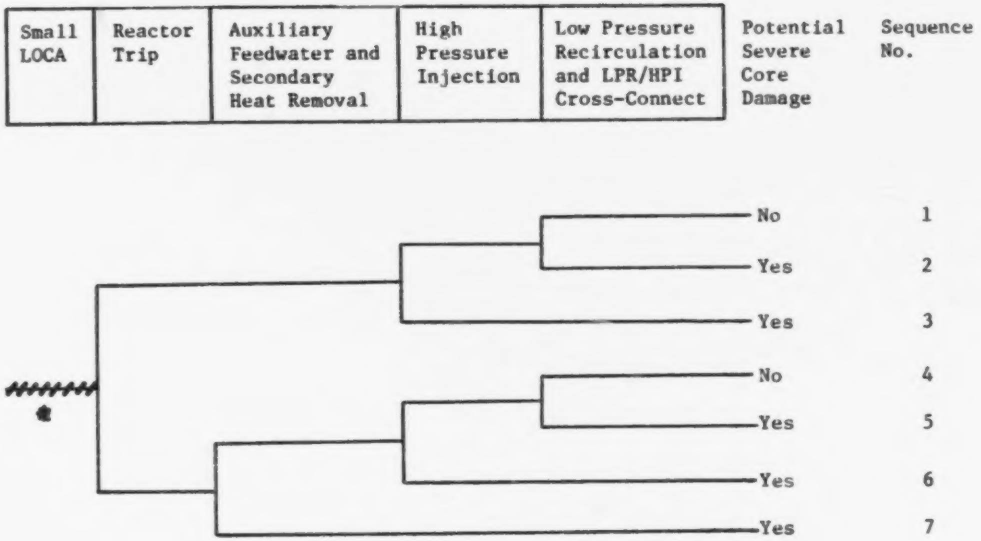
Unavailability of component: PORV failure to close: 10^{-2}/D

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Operator initiated secondary side transient while at 15% power
Pressurizer pilot operated relief valve demanded and opens
Pilot operated relief valve closes
Reactor trip on low pressure
HPI actuation
Relief valve block valve closed by operator
Potential Severe Core Damage
No
No
No
No
Yes
ATWS
No

NSIC 103892 — Actual Occurrence for Excessive Cooldown Rate at Oconee 3


Small LOCA
Reactor Trip
Auxiliary Feedwater and Secondary Heat Removal
High Pressure Injection
Low Pressure Recirculation and LPR/HPI Cross-Connect
Potential Severe Core Damage
Sequence No.
*
No
1
Yes
2
Yes
3
No
4
Yes
5
Yes
6
Yes
7

NSIC 103892 — Sequence of Interest for Excessive Cooldown Rate at Oconee 3

*failure requires operator error in failing to close the PORV isolation valve.

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 103892

DATE OF LER: June 27, 1975

DATE OF EVENT: June 13, 1975

SYSTEM INVOLVED: RCS

COMPONENT INVOLVED: Pilot operated relief valve

CAUSE: PORV stuck open, operator error

SEQUENCE OF INTEREST: LOCA — small break from stuck open PORV

ACTUAL OCCURRENCE: Stuck open PORV

REACTOR NAME: Oconee 3

DOCKET NUMBER: 50-287

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 887 MWe

REACTOR AGE: 2.5 yr

VENDOR: Babcock & Wilcox

ARCHITECT-ENGINEERS: Duke Power Co.

OPERATORS: Duke Power Co.

LOCATION: 30 miles west of Greenville, SC

DURATION: N/A

PLANT OPERATING CONDITION: at 100% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; (c) made inoperable; [✓ (d)] failed open

DISCOVERY METHOD: Transient while operating

COMMENT: —

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 104087

Date: June 25, 1975

Title: Safety Injection Tank Isolation Valves Closed at Maine Yankee

The failure sequence was:

1. During a reactor heatup, the safety injection tank isolation valves were checked to insure they were open.
2. The operator checking the valves observed they were locked and assumed they were locked open when in reality they were locked closed.
3. During a later routine review of the main control board the valve position discrepancy was detected.

Corrective action:

1. The valves were opened and the entire ECCS valve check list was again completed.

Design purpose of failed system or component:

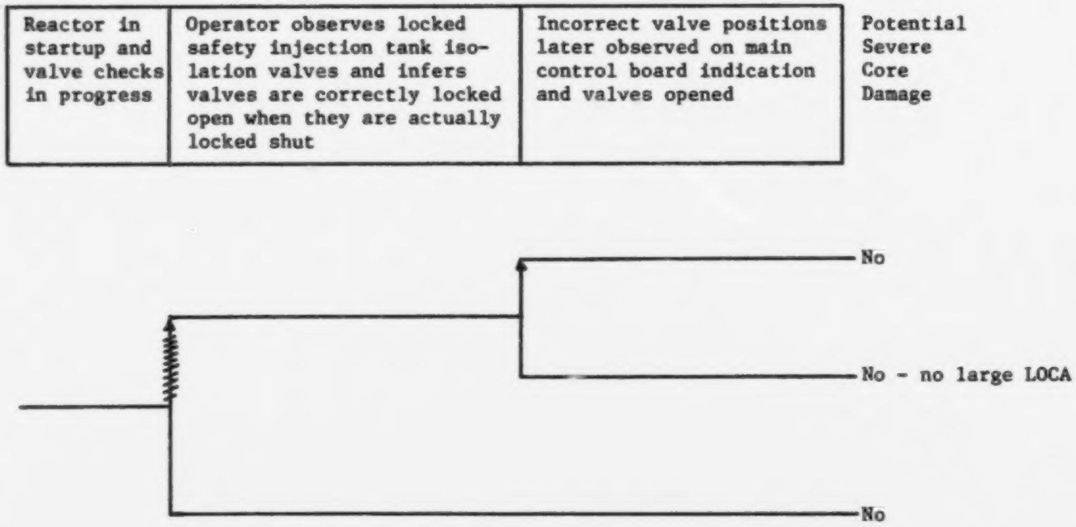
The valves isolate the safety injection tanks from the reactor coolant system when RCS pressure is below safety injection tank pressure.

Unavailability of system per WASH 1400:* Accumulator System: 9.5×10^{-4}/D

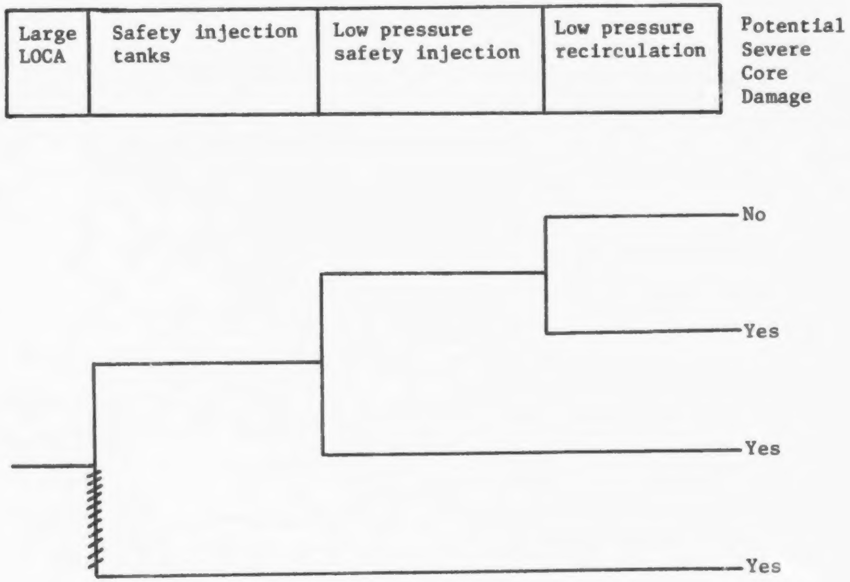
Unavailability of component per WASH 1400:* Operator error (error of commission): 3×10^{-3}/D

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor in startup and valve checks in progress
Operator observes locked safety injection tank isolation valves and infers valves are correctly locked open when they are actually locked shut
Incorrect valve positions later observed on main control board indication and valves opened
Potential Severe Core Damage
No
No - no large LOCA
No

NSIC 104087 — Actual Occurrence for Safety Injection Tank Isolation Valves Closed at Maine Yankee


Large LOCA
Safety injection tanks
Low pressure safety injection
Low pressure recirculation
Potential Severe Core Damage
No
Yes
Yes
Yes

NSIC 104087 — Sequence of Interest for Safety Injection Tank Isolation Valves Closed at Maine Yankee

947

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 104087

DATE OF LER: June 25, 1975

DATE OF EVENT: June 19, 1975

SYSTEM INVOLVED: Safety Injection

COMPONENT INVOLVED: Safety Injection Tanks

CAUSE: Locked Closed SI Tank Isolation Valves, operator error

SEQUENCE OF INTEREST: LOCA (large)

ACTUAL OCCURRENCE: Closed SI tank isolation valves during startup

REACTOR NAME: Maine Yankee

DOCKET NUMBER: 50-309

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 825 MWe

REACTOR AGE: 2.7 yr

VENDOR: Combustion Engineering

ARCHITECT-ENGINEERS: Stone & Webster

OPERATORS: Maine Yankee Atomic Power Co.

LOCATION: 10 miles north of Bath, Maine

DURATION: 4 hours

PLANT OPERATING CONDITION: startup, 0% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________________

DISCOVERY METHOD: operator observation

COMMENT:

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 104379

Date: July 3, 1975

Title: Earthquake Occurs at Humboldt Bay

The failure sequence was:

1. Humboldt Bay was shutdown for refueling.
2. An earthquake of .36 g occurred at the plant.
3. The plant was inspected and no damage was identified.

Corrective action:

None

Design purpose of failed system or component:

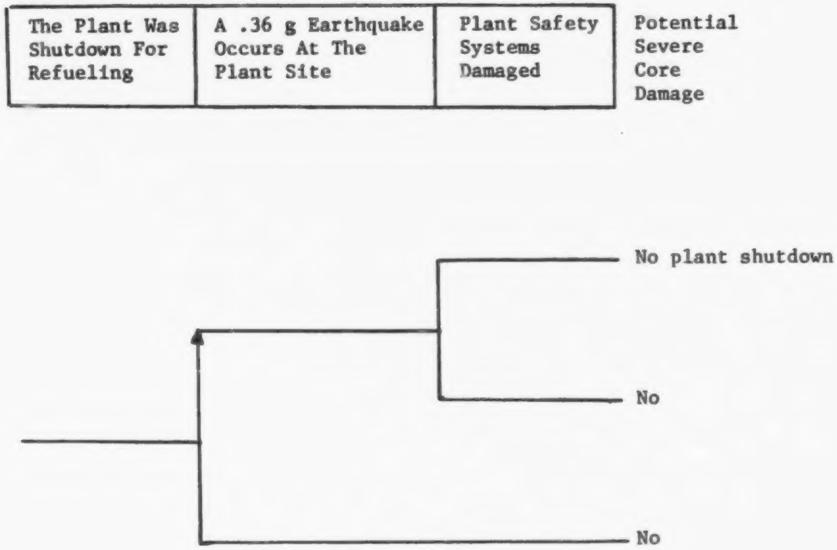
N.A.

Unavailability of system per WASH 1400:* -

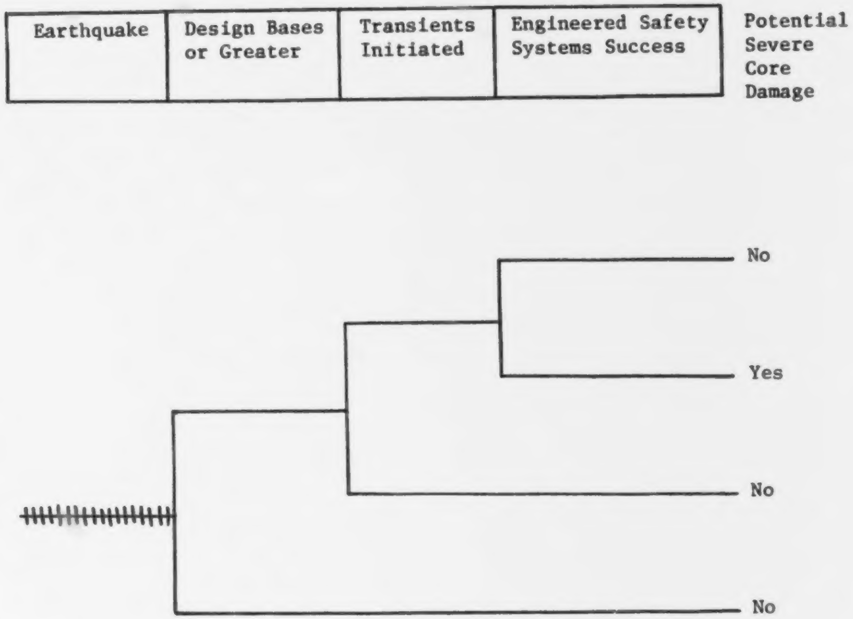
Unavailability of component per WASH 1400:* -

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


The Plant Was Shutdown For Refueling
A .36 g Earthquake Occurs At The Plant Site
Plant Safety Systems Damaged
Potential Severe Core Damage
No plant shutdown
No
No

NSIC 104379 — Actual Occurrence of Earthquake Occurs at Humboldt Bay


Earthquake
Design Bases or Greater
Transients Initiated
Engineered Safety Systems Success
Potential Severe Core Damage
No
Yes
No
No

NSIC 104379 — Sequence of Interest for Earthquake Occurrence at Humboldt Bay

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 104379

DATE OF LER: July 3, 1975

DATE OF EVENT: June 7, 1975

SYSTEM INVOLVED: N.A.

COMPONENT INVOLVED: N.A.

CAUSE: Earthquake

SEQUENCE OF INTEREST: Unique

ACTUAL OCCURRENCE: Earthquake occurs at Humboldt Bay

REACTOR NAME: Humboldt Bay

DOCKET NUMBER: 50-133

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 50 MWe

REACTOR AGE: 12.4 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Pacific Gas & Electric

LOCATION: Four miles South of Eureka

DURATION: N/A

PLANT OPERATING CONDITION: shutdown

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; (c) made inoperable; (d) ____________________

DISCOVERY METHOD: plant transient

COMMENT: Operating basis is .25 g, the acceleration of experiment by the plant was .36 g.

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 104817

Date: July 29, 1975

Title: All ADS Valves Inadvertantly Opened at Hatch 1

The failure sequence was:

1. The HPCI system was found to be inoperable. This required surveillance on the ADS system.
2. Due to improper use of procedures all the ADS valves were inadvertently opened for 90 seconds.
3. The valves were shut by the operator and RCIC was used to recover the reactor water level.

Corrective action:

1. All relief valve piping was inspected.
2. The procedures were rewritten.

Design purpose of failed system or component:

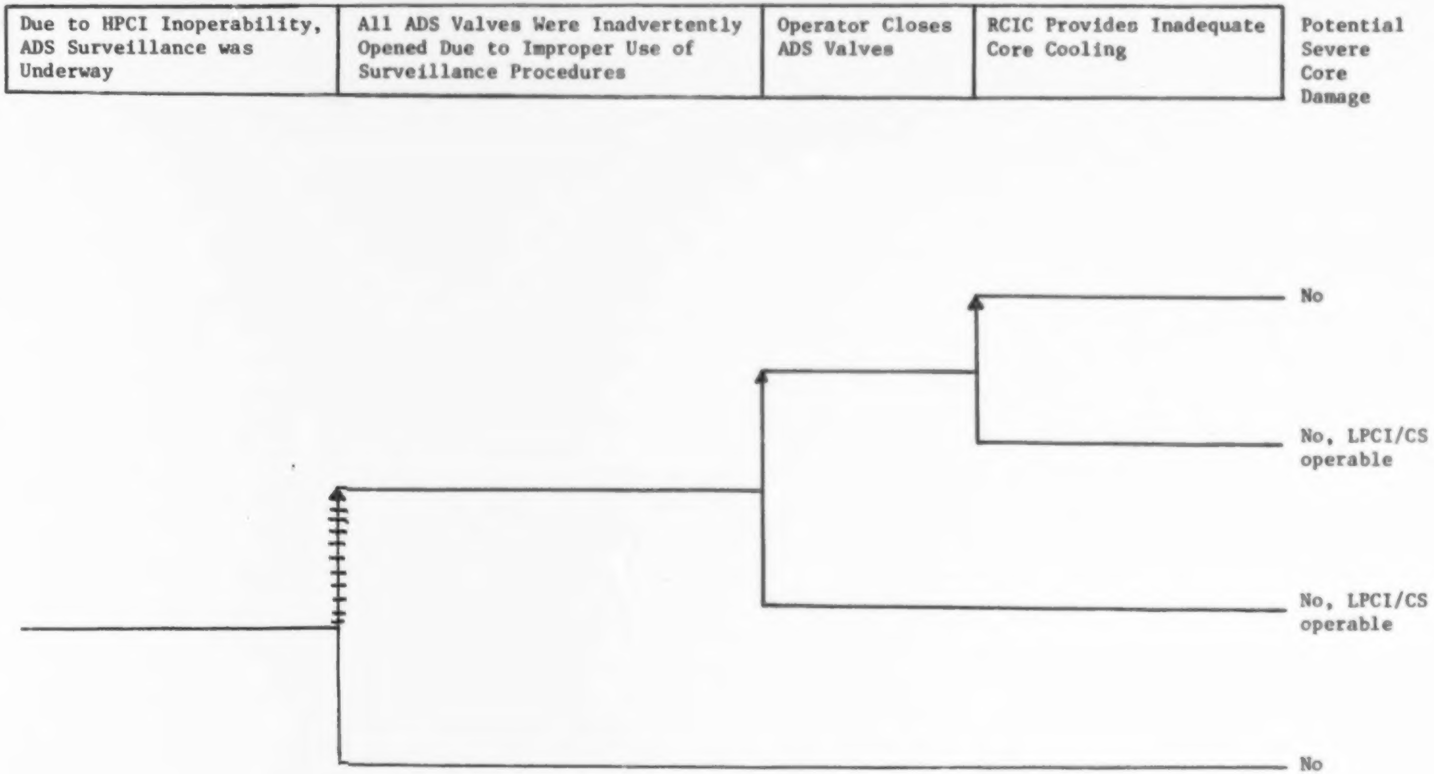
No component was failed.

Unavailability of system per WASH 1400:* -

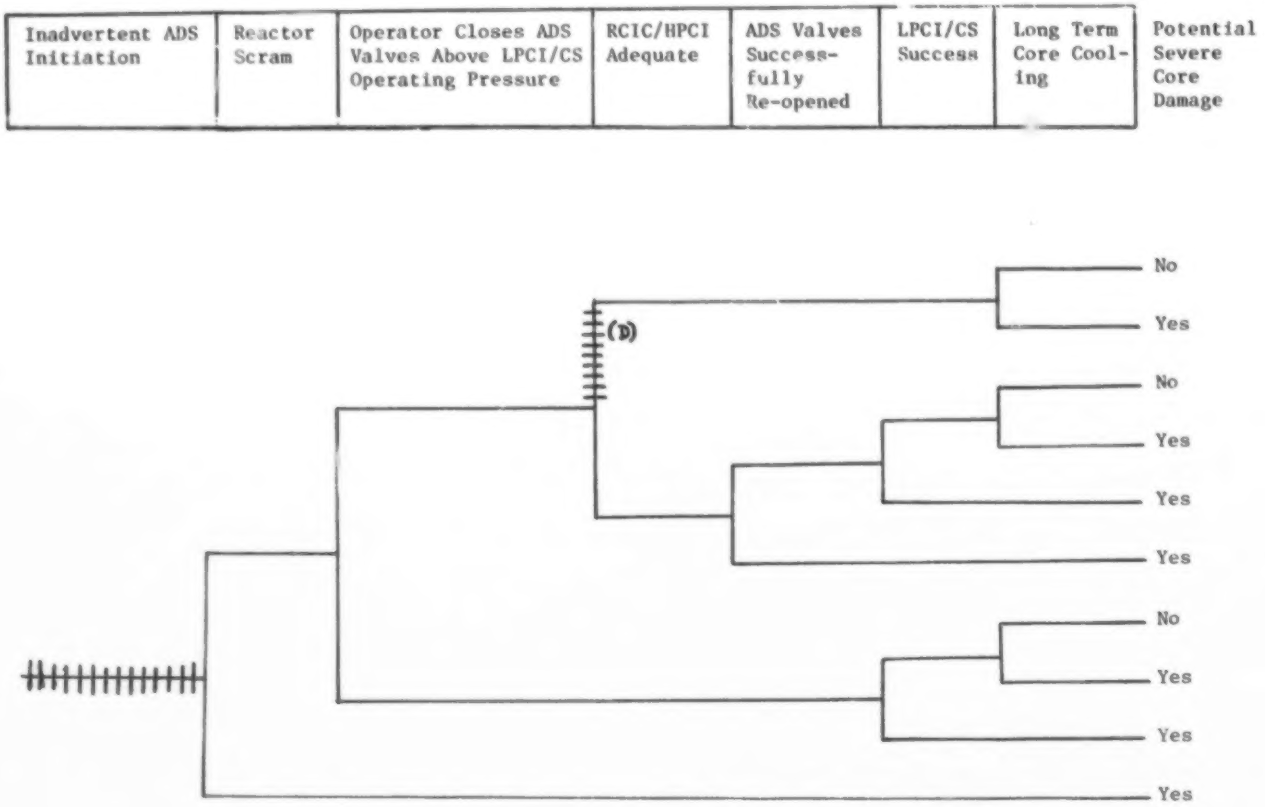
Unavailability of component per WASH 1400:* -

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Due to HPCI Inoperability, ADS Surveillance was Underway
All ADS Valves Were Inadvertently Opened Due to Improper Use of Surveillance Procedures
Operator Closes ADS Valves
RCIC Provides Inadequate Core Cooling
Potential Severe Core Damage
No
No, LPCI/CS operable
No, LPCI/CS operable
No

NSIC 104817 — Actual Occurrence of All ADS Valves Opened Inadvertently at Hatch 1


Inadvertent ADS Initiation
Reactor Scram
Operator Closes ADS Valves Above LPCI/CS Operating Pressure
RCIC/HPCI Adequate
ADS Valves Successfully Re-opened
LPCI/CS Success
Long Term Core Cooling
Potential Severe Core Damage
(D)
No
Yes
No
Yes
Yes
Yes
No
Yes
Yes
Yes

NSIC 104817 — Sequence of Interest for all ADS Valves Opened Inadvertently at Hatch 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 104817

DATE OF LER: August 8, 1975

DATE OF EVENT: July 29, 1975

SYSTEM INVOLVED: ADS

COMPONENT INVOLVED: Valves

CAUSE: Operator Error

SEQUENCE OF INTEREST: Inadvertent ADS Actuation

ACTUAL OCCURRENCE: All ADS valves inadvertently opened

REACTOR NAME: Hatch 1

DOCKET NUMBER: 50-321

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 777 MWe

REACTOR AGE: 0.96 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: Southern Services

OPERATORS: Georgia Power

LOCATION: 11 miles North of Baxley, GA

DURATION: N/A

PLANT OPERATING CONDITION: 84% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; (c) made inoperable; [✓ (d)] inadvertent operation

DISCOVERY METHOD: operator error during surveillance testing

COMMENT:

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: **105540**

Date: **August 17, 1975**

Title: **Low Flow Feedwater Line Severed at 6" to 4" Reducer at Quad Cities**

The failure sequence was:

1. The reactor was increasing power after startup.
2. A transfer was being made from the low flow feed regulating valve to the main feed regulating valve.
3. With both main and low feed regulating valves partially opened, a feedwater vibration alarm was received in the control room.
4. The operator manually scrammed the reactor because the low flow line showed signs that it was starting to sever.
5. The feedwater pumps were tripped and the feedwater regulating station was isolated.
6. Approximately 12,500 gallons of water was spilled; of this 65% was reactor coolant and the remaining 35% was service water.
7. It is believed the failure resulted from vibration of the feedwater regulating station during the transfer.
8. RCIC was used to maintain water level control.

Corrective action:

1. A new 4 x 6 reducer fabricated from more ductile steel was installed.
2. Snubbers were installed to reduce pipe movement.
3. One of the main feedwater valves was replaced with a "drag valve" to provide more control over a wide spectrum of flow conditions.

Design purpose of failed system or component:

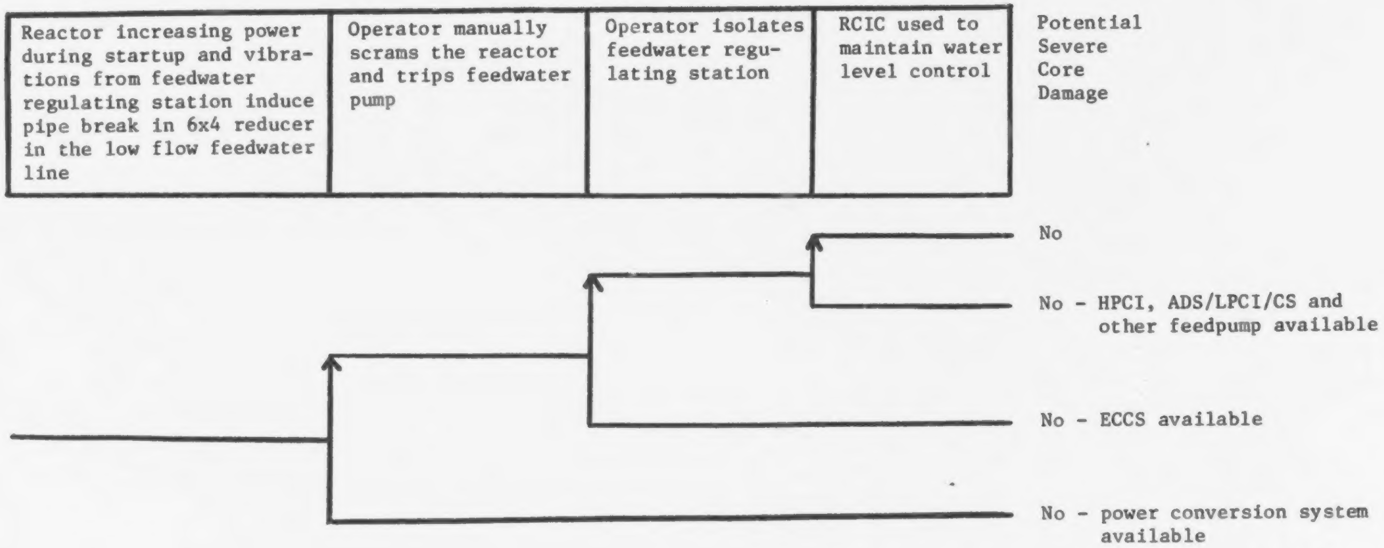
The feedwater system provides feedwater to the reactor.

Unavailability of system per WASH 1400:*

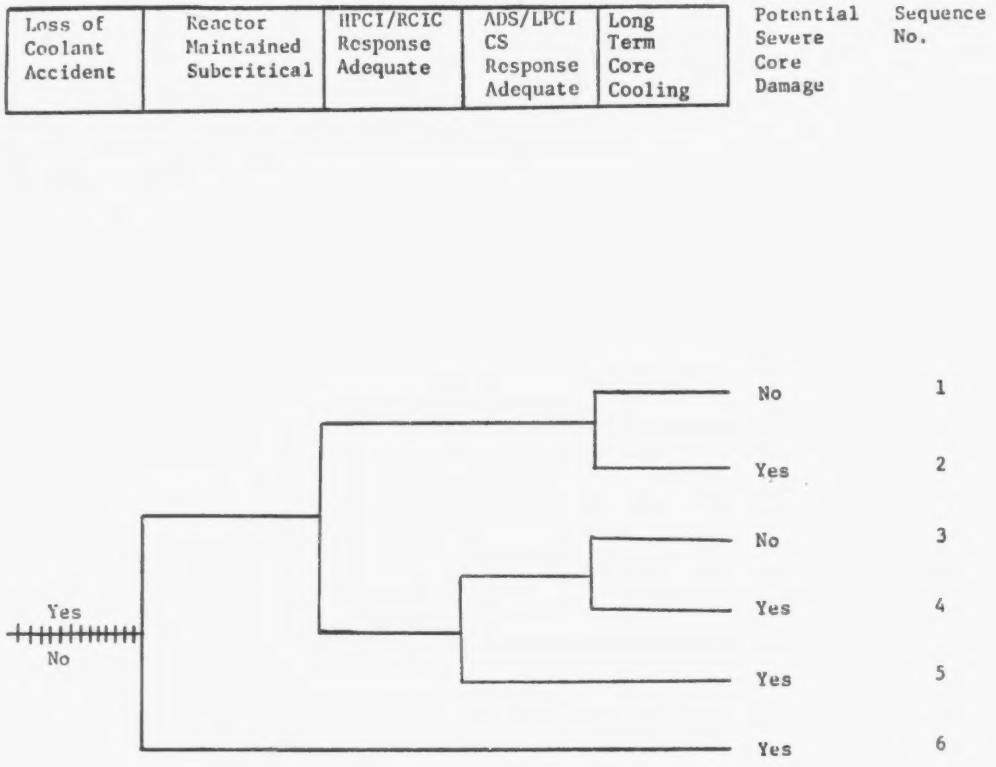
Unavailability of component per WASH 1400:* Pipe >3" 1×10^{-10}/hr.

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor increasing power during startup and vibrations from feedwater regulating station induce pipe break in 6x4 reducer in the low flow feedwater line
Operator manually scrams the reactor and trips feedwater pump
Operator isolates feedwater regulating station
RCIC used to maintain water level control
Potential Severe Core Damage
No
No - HPCI, ADS/LPCI/CS and other feedpump available
No - ECCS available
No - power conversion system available

NSIC 105540 — Actual Occurrence of Low Flow Feedwater Line Severed at 6x4 Reducer at Quad Cities 2


Loss of Coolant Accident
Reactor Maintained Subcritical
HPCI/RCIC Response Adequate
ADS/LPCI CS Response Adequate
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
No
1
Yes
2
No
3
Yes
4
Yes
5
Yes
6

NSIC 105540 — Sequence of Interest for Low Flow Feedwater Line Severed at 6x4 Reducer at Quad Cities 2

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 105540

DATE OF LER: August 27, 1975

DATE OF EVENT: August 17, 1975

SYSTEM INVOLVED: Feedwater

COMPONENT INVOLVED: Piping

CAUSE: Pipe rupture

SEQUENCE OF INTEREST: Loss of Coolant Accident

ACTUAL OCCURRENCE: Low Flow Feedwater Line Severed at 6" to 4" Reducer

REACTOR NAME: Quad Cities 2

DOCKET NUMBER: 50-265

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 789 MWe

REACTOR AGE: 3.3 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: Sargent & Lundy

OPERATORS: Commonwealth Edison

LOCATION: 20 miles NE of Moline, Ill.

DURATION: N/A

PLANT OPERATING CONDITION: Power accension

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; (c) made inoperable; [✓ (d) severed].

DISCOVERY METHOD: Operational Event

COMMENT:

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: **106333**

Date: **August 30, 1975**

Title: **HPCI Found Inoperable at Quad Cities 2**

The failure sequence was:

1. **The RCIC system had failed for reasons unspecified.**
2. **A test was initiated to determine the operability of HPCI**
3. **This test revealed that HPCI was also inoperable due to a broken auxiliary oil pump line.**
4. **Since both RCIC and HPCI were inoperable an orderly shutdown was initiated.**

Corrective action:

1. **The reactor was shutdown in an orderly fashion.**
2. **The HPCI auxiliary oil pump line was replaced with the same type of line.**

Design purpose of failed system or component:

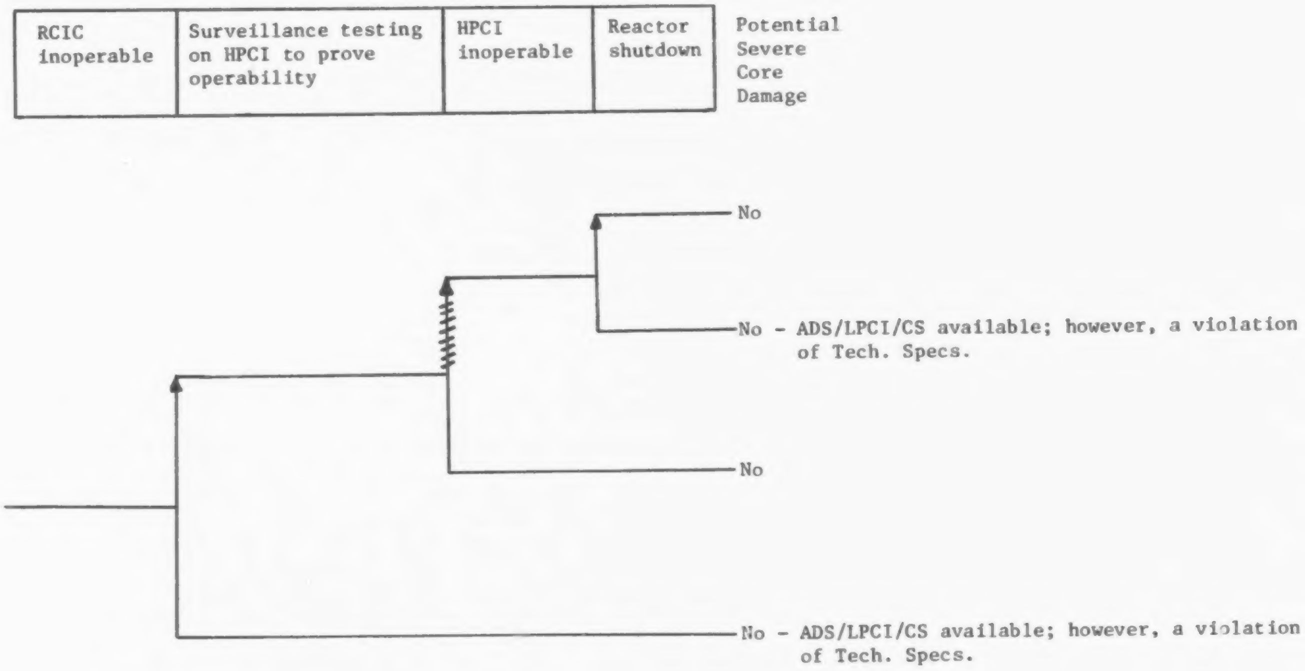
1. **RCIC provides for core cooling when feedwater is unavailable.**
2. **HPCI provides emergency cooling water when the system pressure is above the LPCI and CS operating pressures.**
3. **The auxiliary oil pump provides lubrication to the HPCI turbine prior to initiation of the turbine driven lube oil pump.**

Unavailability of system per WASH 1400:* **HPCI: 8.8×10^{-2}/D**

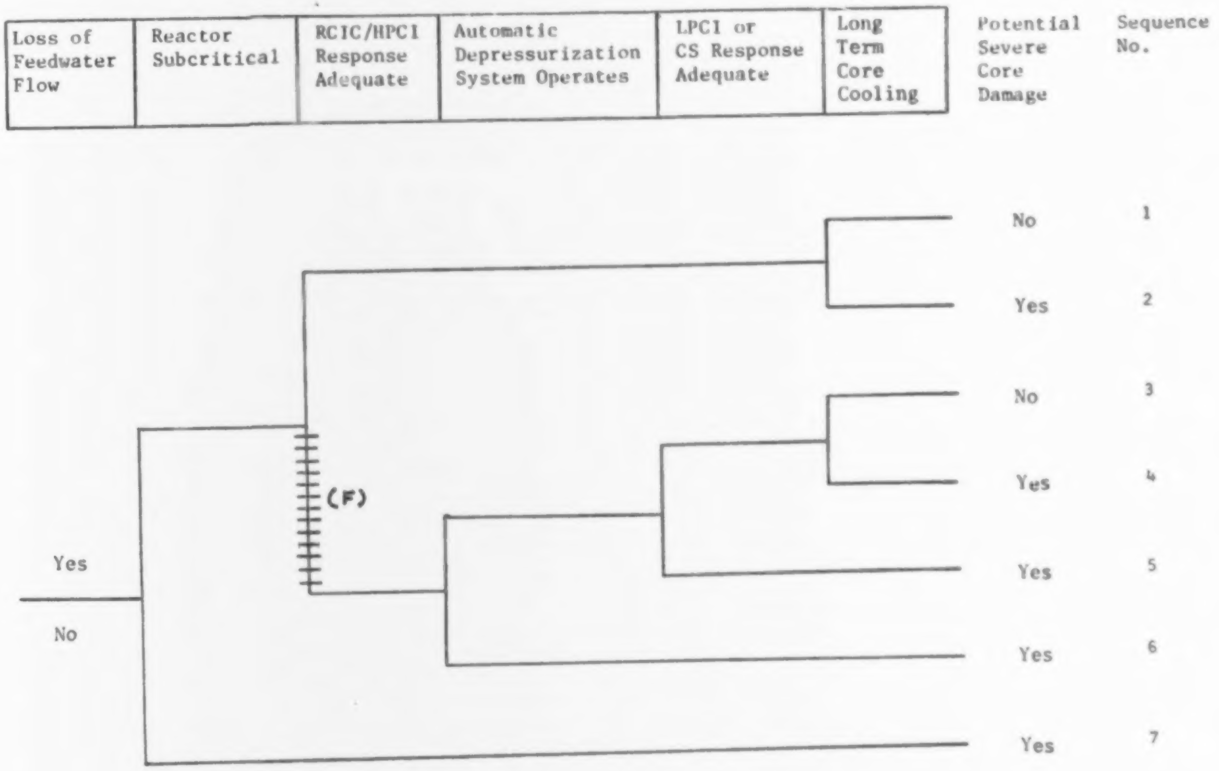
Unavailability of component per WASH 1400:* **pump failure to start: 1×10^{-3}/D**

---

[*] Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


RCIC inoperable
Surveillance testing on HPCI to prove operability
HPCI inoperable
Reactor shutdown
Potential Severe Core Damage
No
No - ADS/LPCI/CS available; however, a violation of Tech. Specs.
No
No - ADS/LPCI/CS available; however, a violation of Tech. Specs.

NSIC 106333 — Actual Occurrence of HPCI Found Inoperable at Quad Cities 2


Loss of Feedwater Flow
Reactor Subcritical
RCIC/HPCI Response Adequate
Automatic Depressurization System Operates
LPCI or CS Response Adequate
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
(F)
No 1
Yes 2
No 3
Yes 4
Yes 5
Yes 6
Yes 7

NSIC 106333 — Sequence of Interest for HPCI Found Inoperable at Quad Cities 2

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 106333

DATE OF LER: September 9, 1975

DATE OF EVENT: August 31, 1975

SYSTEM INVOLVED: HPCI

COMPONENT INVOLVED: aux lube oil pump

CAUSE: broken oil line

SEQUENCE OF INTEREST: loss of feedwater flow

ACTUAL OCCURRENCE: HPCI found inoperable at Quad Cities 2

REACTOR NAME: Quad Cities 2

DOCKET NUMBER: 50-265

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 789 MWe

REACTOR AGE: 3.4 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: Sargent & Lundy

OPERATORS: Commonwealth Edison

LOCATION: 20 miles NE of Moline

DURATION: 360(a) hours

PLANT OPERATING CONDITION: 69% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; [✓ (b)] failed to start;
(c) made inoperable; (d) ______________

DISCOVERY METHOD: surveillance testing

COMMENT: -

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: **106616**

Date: **September 13, 1975**

Title: **Loss of Offsite Power and a Relief Valve Sticks Open at Pilgrim 1**

The failure sequence was:

1. The reactor was undergoing a routine shutdown when a turbine trip occurred (no reason was given for the turbine trip). This caused the startup and auxiliary 345 kv fuses to de-energize (again reason unspecified resulting in a loop). and the reactor to trip.
2. The diesel generators started and re-energized the safety related buses.
3. A relief valve (203-2B) opened for pressure control.
4. This valve stuck open until the reactor pressure fell to approximately 300 psig. The failure was attributed to steam cutting the pilot valve assembly, which increased the pressure on the actuating side of the second stage piston, thereby reducing the closing forces on this piston.

Corrective action:

Corrective action was not discussed in the LER.

Design purpose of failed system or component:

1. The startup buses provide station power when the unit is not generating power.
2. Relief valves provide for pressure control.

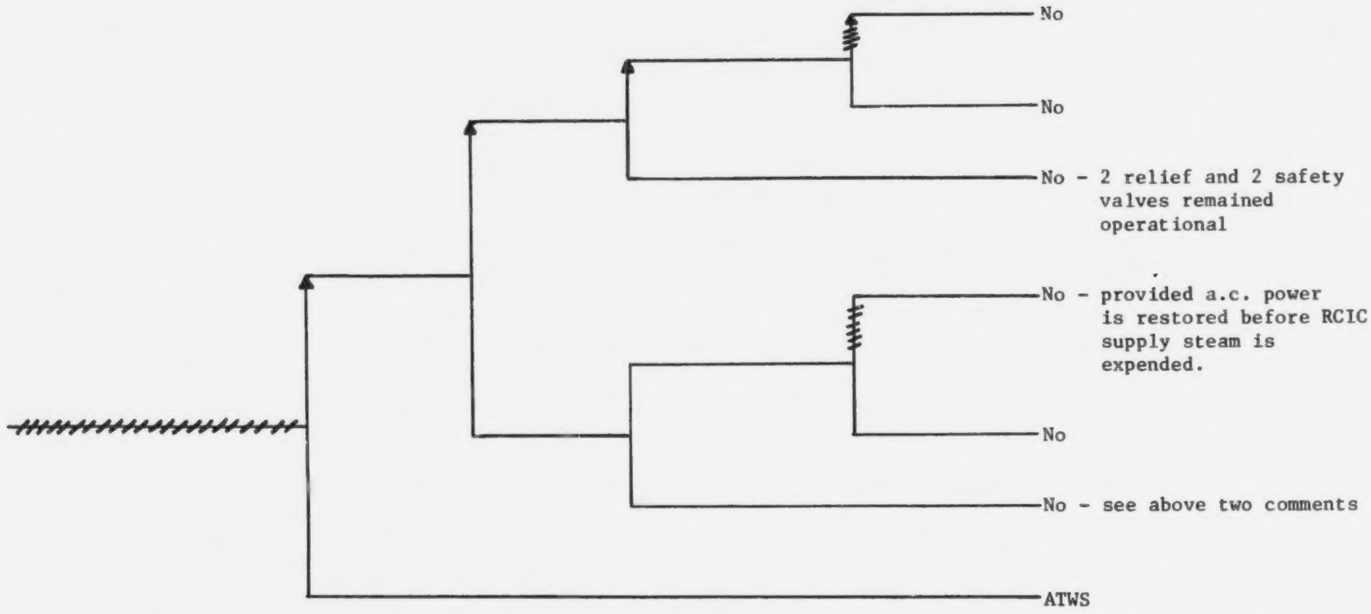
Unavailability of system per WASH 1400:* Loop: 10^{-3}/D following trip.

Unavailability of component per WASH 1400:* PORV failure to close: 10^{-2}/D

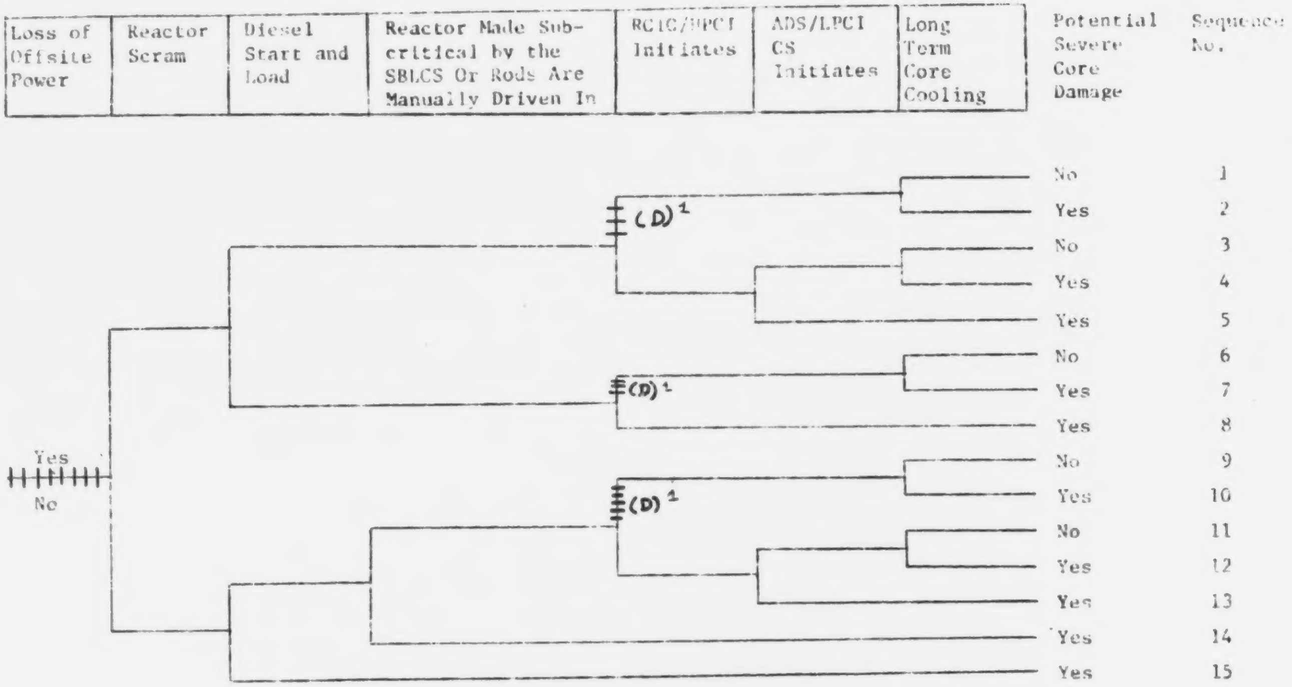
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

As the reactor was experiencing a shutdown, a turbine trip occurred which caused the start-up and auxiliary buses to trip	The reactor scrammed	The diesel generators started and loaded	A relief valve opened for pressure control	The relief valve failed to reseat	Potential Severe Core Damage


No
No
No - 2 relief and 2 safety valves remained operational
No - provided a.c. power is restored before RCIC supply steam is expended.
No
No - see above two comments
ATWS

NSIC 106616 — Actual Occurrence of Loss of Offsite Power and a Relief Valve Sticks Open at Pilgrim 1



Loss of Offsite Power
Reactor Scram
Diesel Start and Load
Reactor Made Sub-critical by the SBLCS Or Rods Are Manually Driven In
RCIC/HPCI Initiates
ADS/LPCI CS Initiates
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
(D)1
(D)1
(D)1
No 1
Yes 2
No 3
Yes 4
Yes 5
No 6
Yes 7
Yes 8
No 9
Yes 10
No 11
Yes 12
Yes 13
Yes 14
Yes 15

NSIC 106616 – Sequence of Interst for Loss of Offsite Power and a Relief Valve Sticks Open at Pilgrim 1

1 A relief valve stuck open.

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 106616

DATE OF LER: September 14, 1975

DATE OF EVENT: September 13, 1975

SYSTEM INVOLVED: electric power, reactor pressure relief system

COMPONENT INVOLVED: startup and auxiliary transfers, relief valve

CAUSE: turbine trip (electric), mechanical failure (relief valve)

SEQUENCE OF INTEREST: loss of offsite power

ACTUAL OCCURRENCE: loss of offsite power and a relief valve sticks open at Pilgrim 1

REACTOR NAME: Pilgrim 1

DOCKET NUMBER: 50-293

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 655 MWe

REACTOR AGE: 3.3 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Boston Edison

LOCATION: 4 miles SE of Plymouth, Mass.

DURATION: N/A

PLANT OPERATING CONDITION: undergoing a routine shutdown

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; [✓ (d)] degraded

DISCOVERY METHOD: operational event

COMMENT: -

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 108078

Date: November 14, 1975

Title: Auxiliary Feedwater Pump Flows Restricted at Kewaunee

The failure sequence was:

1. Resin beads were leaking from a mixed-bed demineralizer in the makeup water system and travelling to the condensate storage tank.
2. During a unit startup, AFW pumps 1A and 1B suction strainers became clogged with resin beads, resulting in low pump flow.
3. The turbine-driven AFW pump (1C) suction strainer was also clogged, which resulted in low flow from that pump also.

Corrective action:

1. The condensate storage tank and the pump suction strainers were cleaned and returned to service.
2. At a later date the fine mesh strainers were to have been removed.

Design purpose of failed system or component:

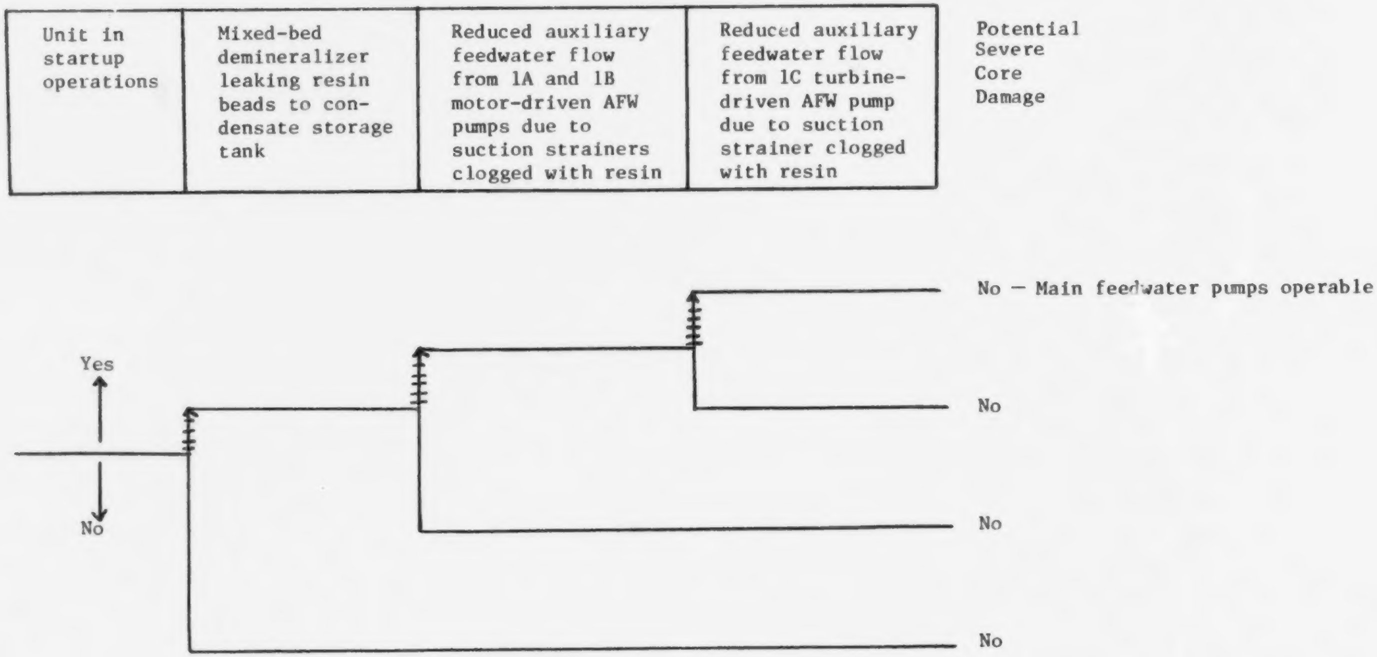
1. The auxiliary feedwater pumps provide water to the steam generators for reactor cooling in the event the main feedwater system is not operable.

Unavailability of system per WASH 1400:* auxiliary feedwater: 2.5×10^{-4}
(start + 8 hours)

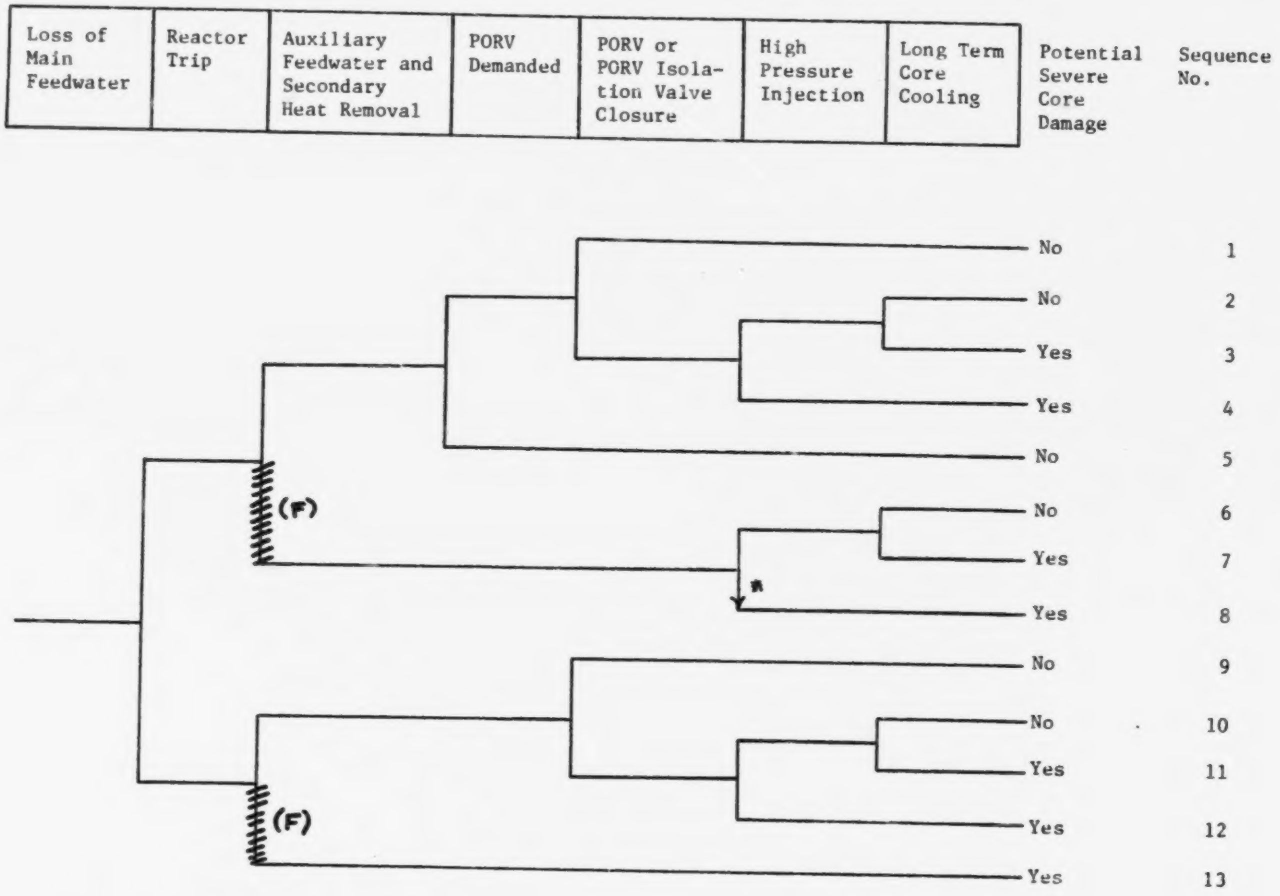
Unavailability of component per WASH 1400:* strainers not included
pumps, failure to start: 1×10^{-3}/D
pumps, failure to run: 3×10^{-5}/hr

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Unit in startup operations
Mixed-bed demineralizer leaking resin beads to condensate storage tank
Reduced auxiliary feedwater flow from 1A and 1B motor-driven AFW pumps due to suction strainers clogged with resin
Reduced auxiliary feedwater flow from 1C turbine-driven AFW pump due to suction strainer clogged with resin
Potential Severe Core Damage
Yes
No
No — Main feedwater pumps operable
No
No
No

NSIC 108078 — Actual Occurrence for Auxiliary Feedwater Pump Flows Restricted at Kewaunee


Loss of Main Feedwater
Reactor Trip
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
(F)
*
(F)
No 1
No 2
Yes 3
Yes 4
No 5
No 6
Yes 7
Yes 8
No 9
No 10
Yes 11
Yes 12
Yes 13

NSIC 108078 — Sequence of Interest for Auxiliary Feedwater Pump Flows Restricted at Kewaunee

*not included in mitigation procedures

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 108078

DATE OF LER: November 14, 1975

DATE OF EVENT: November 5, 1975

SYSTEM INVOLVED: Auxiliary Feedwater

COMPONENT INVOLVED: AFW Pump Strainers

CAUSE: Strainers clogged with resin beads

SEQUENCE OF INTEREST: Loss of feedwater

ACTUAL OCCURRENCE: Inoperable AFW pumps during plant startup

REACTOR NAME: Kewaunee

DOCKET NUMBER: 50-305

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 535 MWe

REACTOR AGE: 1.9 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Pioneer Services

OPERATORS: Wisconsin Public Service Corp.

LOCATION: 27 miles east of Greenbay, Wisc.

DURATION: 360(a) hours

PLANT OPERATING CONDITION: Plant startup

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ________________

DISCOVERY METHOD: during operation

COMMENT: –

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 108869

Date: December 20, 1975

Title: Two Diesel Generators Fails to Start at Hatch 1

The failure sequence was:

1. During a weekly performance run of the diesel generators, 1B and 1C failed to start.
2. Another start test was performed after 24 hours and all diesels started.
3. The cause of the diesels failure to start was not determined, nor was it stated why the restart test was conducted 24 hours later.

Corrective action:

There was no corrective action.

Design purpose of failed system or component:

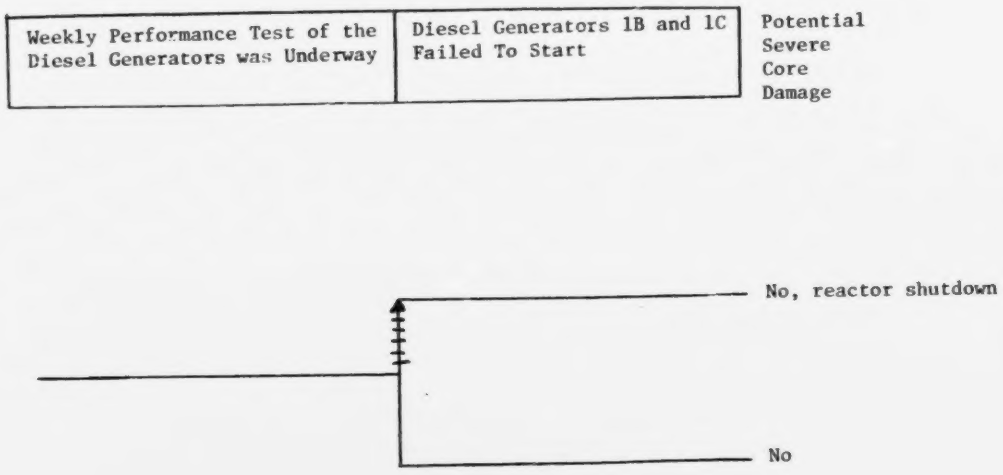
The diesel generators provide a.c. power when offsite sources are unavailable.

Unavailability of system per WASH 1400:* Emergency Power: 10^{-3}/D

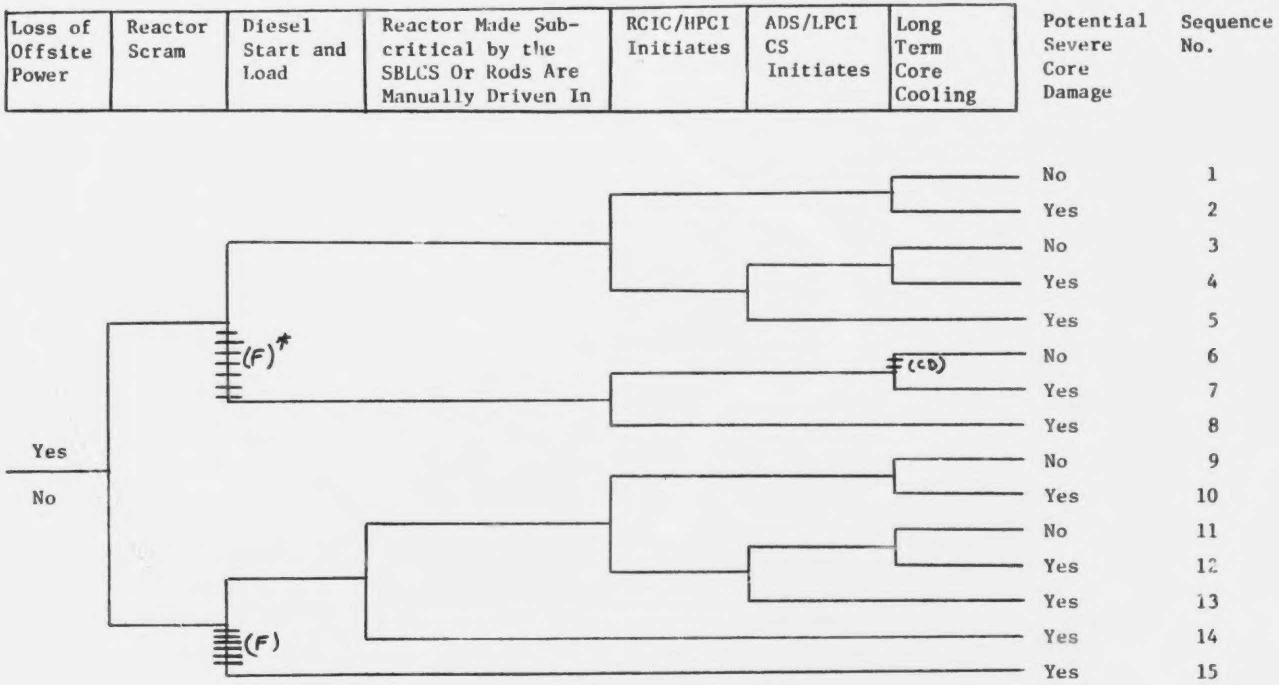
Unavailability of component per WASH 1400:* Diesel Generator failure to start: 3×10^{-2}/D

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Weekly Performance Test of the Diesel Generators was Underway
Diesel Generators 1B and 1C Failed To Start
Potential Severe Core Damage
No, reactor shutdown
No

NSIC 108869 — Actual Occurrence of Two Diesel Generators Fail to Start at Hatch 1


Loss of Offsite Power
Reactor Scram
Diesel Start and Load
Reactor Made Sub-critical by the SBLCS Or Rods Are Manually Driven In
RCIC/HPCI Initiates
ADS/LPCI CS Initiates
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
(F)*
(CD)
(F)
No 1
Yes 2
No 3
Yes 4
Yes 5
No 6
Yes 7
Yes 8
No 9
Yes 10
No 11
Yes 12
Yes 13
Yes 14
Yes 15

NSIC 108869 – Sequence of Interest for Two Diesel Generators Fail to Start at Hatch 1

*Proper load shedding by the operator may result in a degraded state of the emergency power system.

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 108869

DATE OF LER: December 30, 1975

DATE OF EVENT: December 20, 1975

SYSTEM INVOLVED: Electric Power

COMPONENT INVOLVED: Diesel Generators

CAUSE: not given

SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE: Two diesel generators fail to start

REACTOR NAME: Hatch 1

DOCKET NUMBER: 50-321

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 786 MWe

REACTOR AGE: 1.3 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: Southern Services

OPERATORS: Georgia Power

LOCATION: 11 miles North of Baxley, GA

DURATION: 84 (a) hours

PLANT OPERATING CONDITION: 0%

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; [✓ (b)] failed to start;
(c) made inoperable; (d) ______________________

DISCOVERY METHOD: Testing

COMMENT: All diesels started after waiting 24 hours.

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 109295

Date: December 19, 1975

Title: Borated Water Storage Tank Level Indication Fails at Oconee

The failure sequence was:

1. Both channels of BWST level indication failed due to freezing of moisture in the level transmitter impulse lines. This was caused by a failure in the trace heating for the instrument air lines to the transmitters.

Corrective action;

1. Reactor shutdown was initiated.
2. Independent power sources were provided for each heat tracing circuit, resulting in redundant heat tracing.

Design purpose of failed system or component:

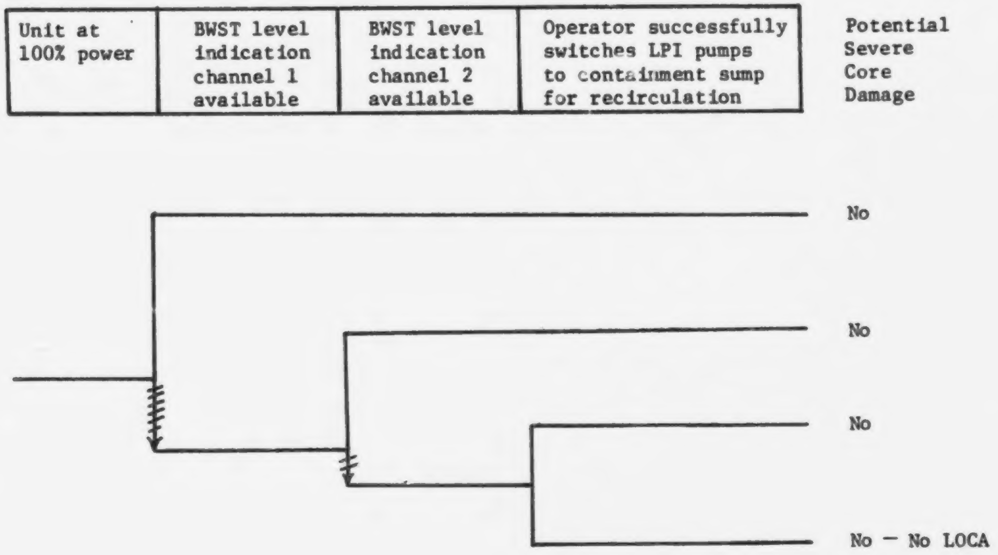
1. The level transmitters provide alarms when the BWST level is such that the operator should switch suction of the emergency core cooling pumps from the BWST to the containment sump for recirculation cooling.

Unavailability of system per WASH 1400:* low pressure recirculation: 3.1×10^{-2}

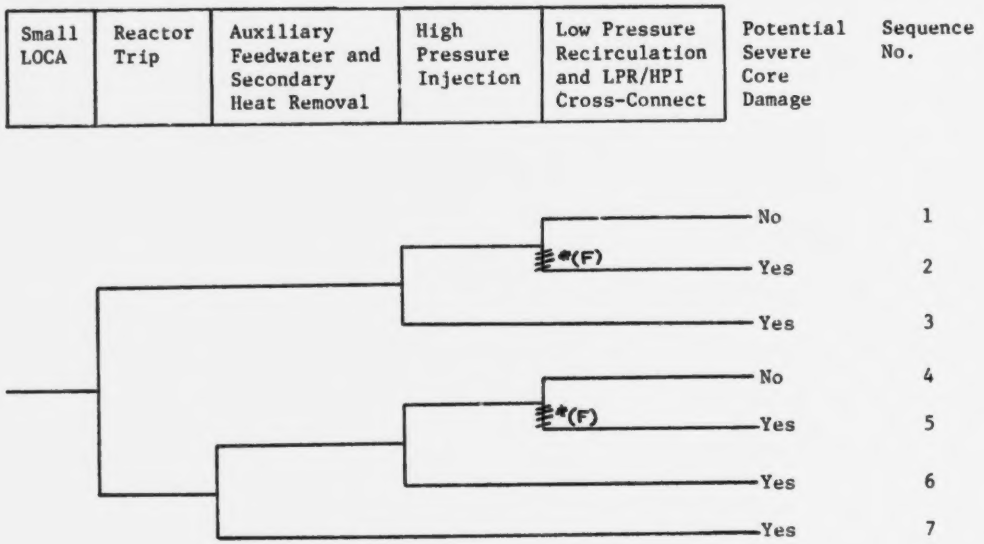
Unavailability of component per WASH 1400:* Failure rate of wires to open: 3×10^{-6}/hr.

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Unit at 100% power
BWST level indication channel 1 available
BWST level indication channel 2 available
Operator successfully switches LPI pumps to containment sump for recirculation
Potential Severe Core Damage
No
No
No
No — No LOCA

NSIC 109295 — Actual Occurrence for Failure to BWST Level Indication at Oconee 1


Small LOCA
Reactor Trip
Auxiliary Feedwater and Secondary Heat Removal
High Pressure Injection
Low Pressure Recirculation and LPR/HPI Cross-Connect
Potential Severe Core Damage
Sequence No.
*(F)
*(F)
No 1
Yes 2
Yes 3
No 4
Yes 5
Yes 6
Yes 7

NSIC 109295 – Sequence of Interest for Failure to BWST Level Indication at Oconee 1

*success requires the operator to switch to recirculation without BWST indication after sufficient water has been injected for LP pump NPSH and before the BWST is emptied, and without level indication.

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 109295

DATE OF LER: January 7, 1976

DATE OF EVENT: December 19, 1975

SYSTEM INVOLVED: Low pressure injection (decay heat removal)

COMPONENT INVOLVED: BWST level indicators

CAUSE: Frozen impulse line due to heat tracing failure

SEQUENCE OF INTEREST: LOCA

ACTUAL OCCURRENCE: Frozen BWST level impulse lines during power operation

REACTOR NAME: Oconee 1

DOCKET NUMBER: 50-269

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 887 MWe

REACTOR AGE: 3.0 yr

VENDOR: B&W

ARCHITECT-ENGINEERS: Duke Power Co.

OPERATORS: Duke Power Co.

LOCATION: 30 miles west of Greenville, SC

DURATION: 360(a) hours

PLANT OPERATING CONDITION: at 100% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ________________

DISCOVERY METHOD: During power operation

COMMENT: –

BEST DOCUMENT AVAILABLE

280

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 112309

Date: March 8, 1976

Title: Gas Turbine Becomes Inoperable at Millstone

The failure sequence was:

1. The isolation condensor was inoperable.
2. It was decided to start the emergency power system (a gas turbine and a diesel generator) as a precaution as hurricane Belle passed.
3. The gas turbine failed to run because the governor was out of adjustment.
4. The reactor was placed in cold shutdown since both the IC and the FWCI systems were inoperable.

Corrective action:

The governor was adjusted.

Design purpose of failed system or component:

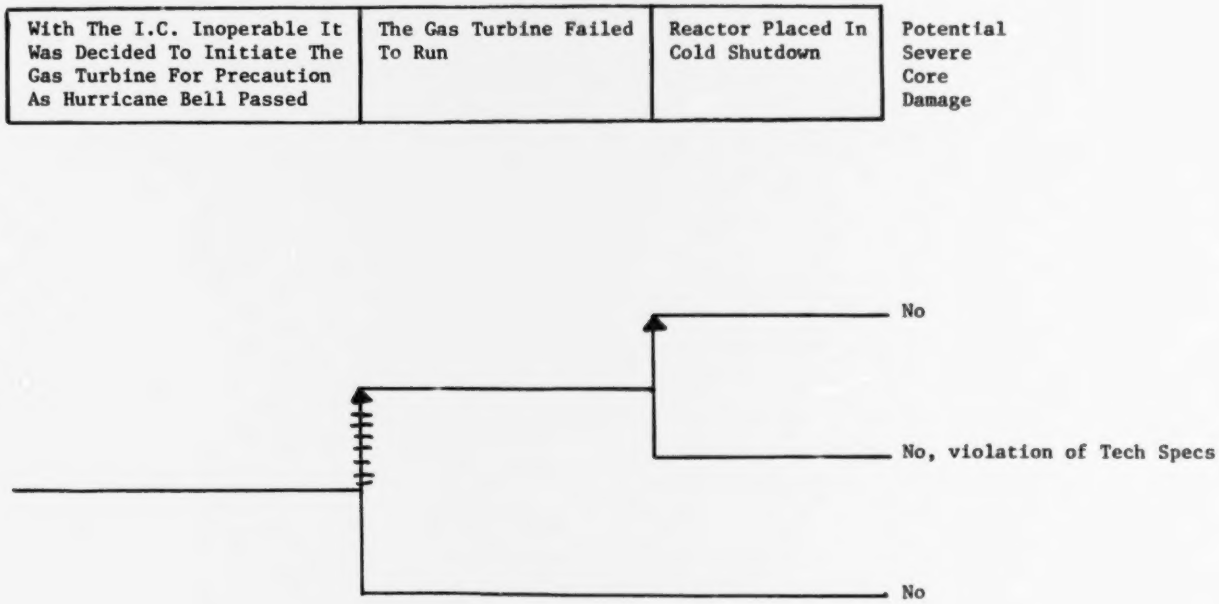
1. The I.C. is a natural circulation system which serves as a heat sink when the reactor is isolated.
2. The FWCI system provides for a source of high pressure coolant given a Loca.

Unavailability of system per WASH 1400:* —

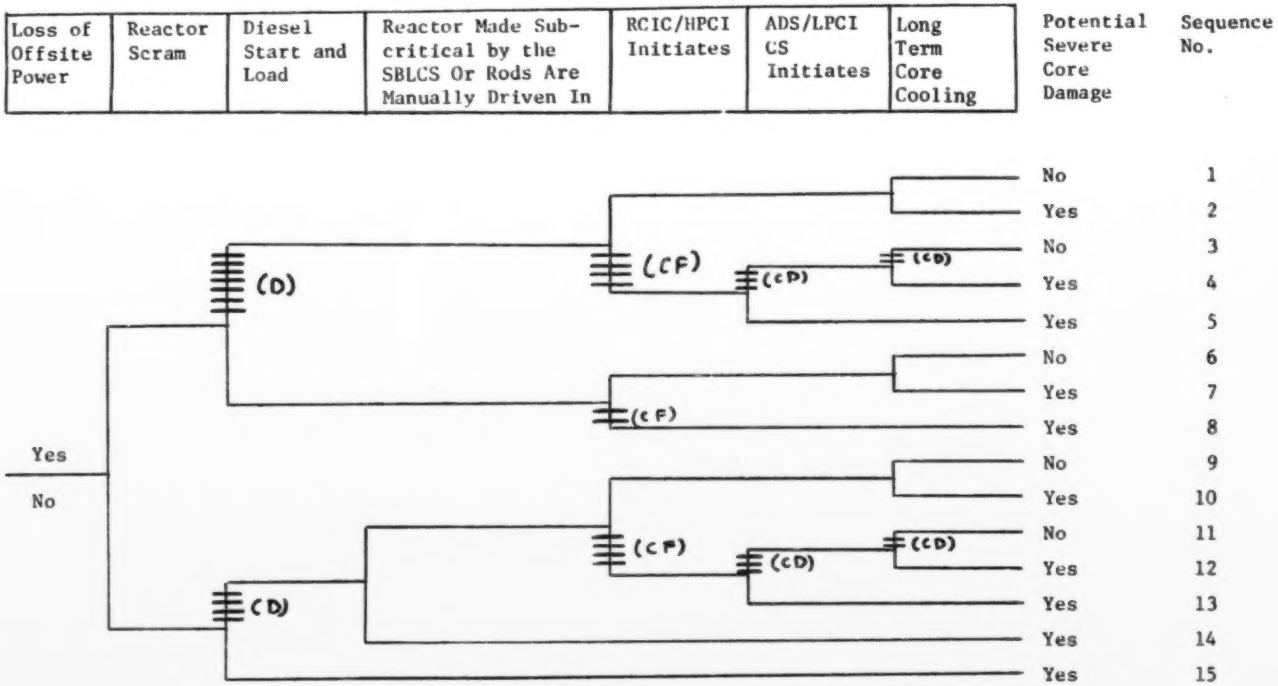
Unavailability of component per WASH 1400:* —

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


With The I.C. Inoperable It Was Decided To Initiate The Gas Turbine For Precaution As Hurricane Bell Passed
The Gas Turbine Failed To Run
Reactor Placed In Cold Shutdown
Potential Severe Core Damage
No
No, violation of Tech Specs
No

NSIC 112309 – Actual Occurrence of Gas Turbine Becomes Inoperable at Millstone 1


Loss of Offsite Power
Reactor Scram
Diesel Start and Load
Reactor Made Subcritical by the SBLCS Or Rods Are Manually Driven In
RCIC/HPCI Initiates
ADS/LPCI CS Initiates
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
(D)
(CF)
(CD)
(CD)
(CF)
(CF)
(CD)
(CD)
(D)
No 1
Yes 2
No 3
Yes 4
Yes 5
No 6
Yes 7
Yes 8
No 9
Yes 10
No 11
Yes 12
Yes 13
Yes 14
Yes 15

NSIC 112309 — Sequence of Interest of Gas Turbine Becomes Inoperable at Millstone 1

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 112309

DATE OF LER: March 22, 1976

DATE OF EVENT: March 8, 1976

SYSTEM INVOLVED: I.C., FWCI, Emergency Power

COMPONENT INVOLVED: Governor

CAUSE: Out of adjustment

SEQUENCE OF INTEREST: Loss of offsite power.

ACTUAL OCCURRENCE: Gas Turbine Becomes Inoperable

REACTOR NAME: Millstone 1

DOCKET NUMBER: 245

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 660 MWe

REACTOR AGE: 5.4 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: Ebasco

OPERATORS: North East Nuclear Energy Company

LOCATION: Five miles SW of New London

DURATION: 12 (a) hours

PLANT OPERATING CONDITION: 100% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________________

DISCOVERY METHOD: Operational Event.

COMMENT:

BEST DOCUMENT AVAILABLE 244

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 112310

Date: March 15, 1976

Title: Gas Turbine Fails to Achieve Rated Speed at Millstone 1

The failure sequence was:

1. The gas turbine was being tested daily since the isolation condensor (IC) was inoperable. The feedwater coolant injection system, which is redundant to the IC, requires that the gas turbine be operable given a loss of offsite power.
2. During the surveillance test the gas turbine initiated that failed to reach rated speed. Electronic components in the governor were malfunctioning.
3. The unit was placed in cold shutdown.

Corrective action:

The defective electronic components were replaced.

Design purpose of failed system or component:

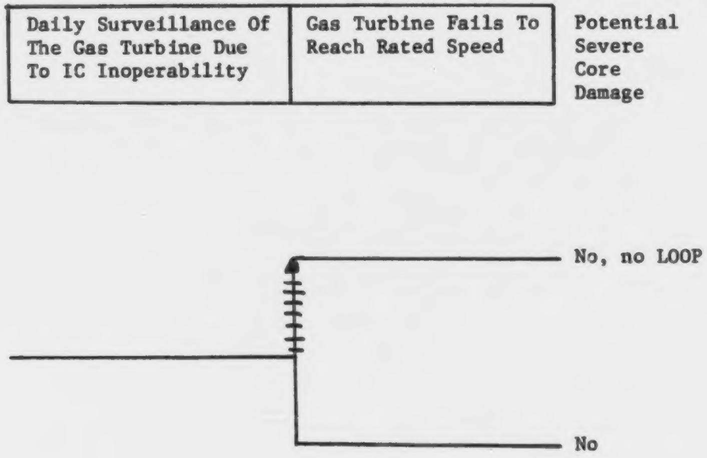
1. The IC is a natural circulation heat sink used when the reactor is isolated.
2. The feedwater coolant injection system provides a high pressure source of water given a small LOCA.

Unavailability of system per WASH 1400:[*] IC: N.A.
FWCI: N.A.

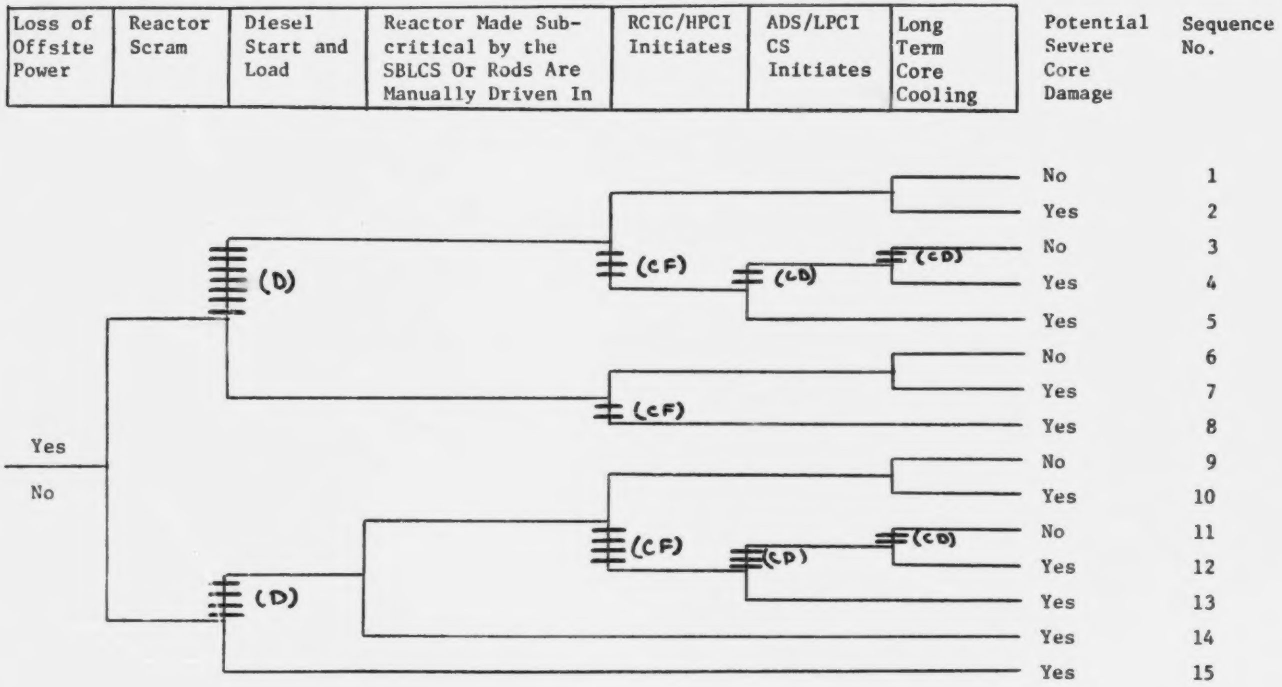
Unavailability of component per WASH 1400:[*] Gas Turbine: N.A.
Electronic components (solid state devices, low power application):
3.10^{-6}/HR

---

[*] Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Daily Surveillance Of The Gas Turbine Due To IC Inoperability
Gas Turbine Fails To Reach Rated Speed
Potential Severe Core Damage
No, no LOOP
No

NSIC 112310 — Actual Occurrence of Gas Turbine Fails to Achieve Rated Speed at Millstone 1


Loss of Offsite Power
Reactor Scram
Diesel Start and Load
Reactor Made Subcritical by the SBLCS Or Rods Are Manually Driven In
RCIC/HPCI Initiates
ADS/LPCI CS Initiates
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
(D)
(CF)
(CD)
(CD)
(CF)
(D)
(CF)
(CD)
(CD)
No 1
Yes 2
No 3
Yes 4
Yes 5
No 6
Yes 7
Yes 8
No 9
Yes 10
No 11
Yes 12
Yes 13
Yes 14
Yes 15

NSIC 112310 - Sequence of Interest for Gas Turbine Fails to Achieve Rated Speed at Millstone 1

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 112310

DATE OF LER: March 29, 1976

DATE OF EVENT: March 15, 1976

SYSTEM INVOLVED: I.C., FWCI, Emergency Power.

COMPONENT INVOLVED: Gas turbine

CAUSE: Failure of an electronic component

SEQUENCE OF INTEREST: Loss of offsite power.

ACTUAL OCCURRENCE: Gas Turbine Fails to Achieve Rated Speed.

REACTOR NAME: Millstone - 1

DOCKET NUMBER: 245

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 660 MWe

REACTOR AGE: 5.4 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: Ebasco.

OPERATORS: Northeastern Nuclear Energy Company

LOCATION: Five miles SW of New London

DURATION: 12 (a) hours

PLANT OPERATING CONDITION: At Power.

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________________

DISCOVERY METHOD: Surveillence testing.

COMMENT:

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 115870

Date: July 6, 1976

Title: Main Steam Relief Valve Fails to Operate at Vermont Yankee

The failure sequence was:

1. During surveillance testing it was noted that the air operator plunger on a relief valve (S/N 12) did not move when air was applied.
2. It was decided to test all relief valves. Of the remaining three, two additional valves failed (S/N 14 and S/N 37).
3. The valve air operator diaphragms had failed due to excessive heating. This was caused in part by (1) improper insulation in the proximity of the diaphragms and (2) an extended operating cycle.

Corrective action:

1. The diaphragms were replaced.
2. A modification (not specified) to the valve insulation was made.
3. The valve operating cycles will be shortened. Operability checks of the air operator diaphragm followed by a diaphragm replacement on all valves and a subsequent retest of the new diaphragms independent of the initial test result will be carried out during each refueling outage.

Design purpose of failed system or component:

1. The ADS system serves as a backup to the HPCI.
2. Relief valves allow for reactor pressure control.

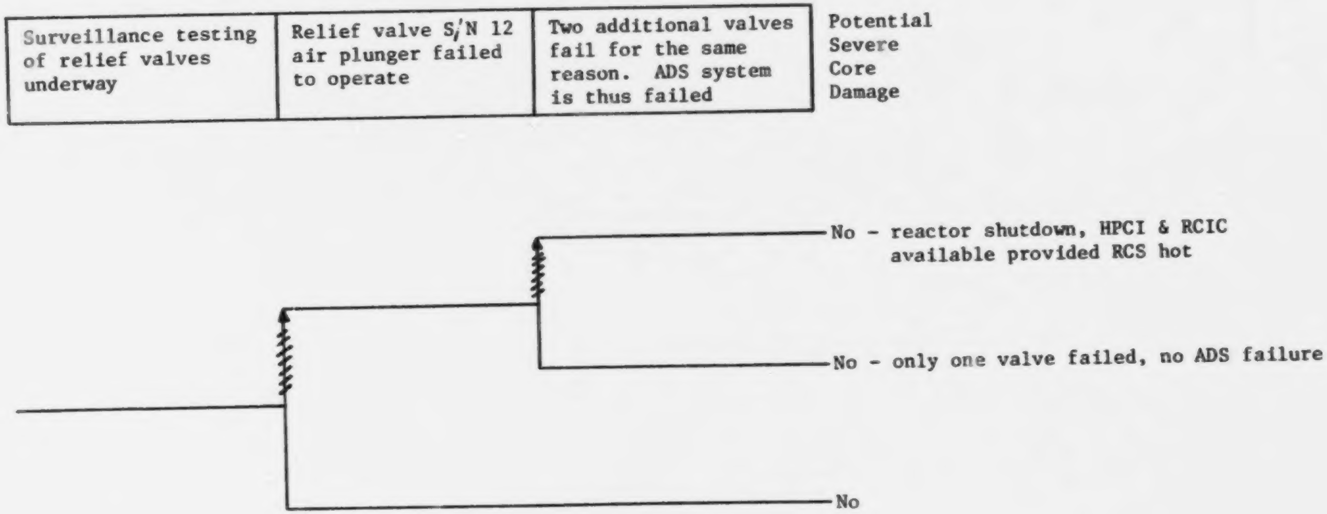
Unavailability of system per WASH 1400:* ADS: $5 \times 10^{-3}/D$

Unavailability of component per WASH 1400:* relief valve failure to open: $1 \times 10^{-5}/D$

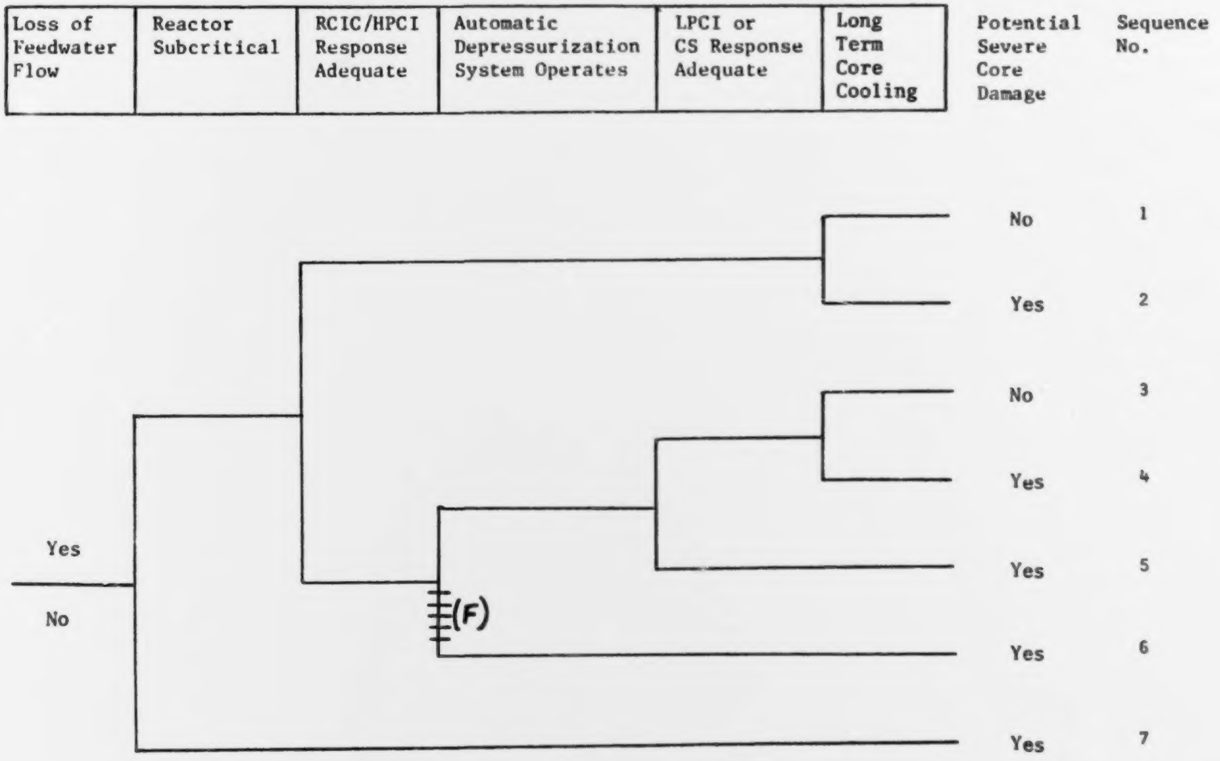
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

BEST DOCUMENT AVAILABLE


Surveillance testing of relief valves underway
Relief valve S/N 12 air plunger failed to operate
Two additional valves fail for the same reason. ADS system is thus failed
Potential Severe Core Damage
No - reactor shutdown, HPCI & RCIC available provided RCS hot
No - only one valve failed, no ADS failure
No

NSIC 115870 — Actual Occurrence for Main Steam Relief Valve Fails to Operate at Vermont Yankee


Loss of Feedwater Flow
Reactor Subcritical
RCIC/HPCI Response Adequate
Automatic Depressurization System Operates
LPCI or CS Response Adequate
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No
1
Yes
2
No
3
Yes
4
Yes
5
Yes
6
Yes
7
Yes
No
(F)

NSIC 115870 — Sequence of Interest for Main Steam Relief Valve Fails to Operate at Vermont Yankee

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 115870

DATE OF LER: July 20, 1976

DATE OF EVENT: July 6, 1976

SYSTEM INVOLVED: ADS

COMPONENT INVOLVED: relief valves

CAUSE: failure of the air operator diaphragms

SEQUENCE OF INTEREST: Loss of feedwater

ACTUAL OCCURRENCE: main steam relief valves fail to operate at Vermont Yankee

REACTOR NAME: Vermont Yankee

DOCKET NUMBER: 50-271

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 514 MWe

REACTOR AGE: 4.3 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: Ebasco

OPERATORS: Vermont Yankee Nuclear Power Corp.

LOCATION: 5 miles S of Battleboro

DURATION: 4383 (a) hours

PLANT OPERATING CONDITION: 0% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ______________________

DISCOVERY METHOD: surveillance testing

COMMENT: -

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 115875

Date: July 9, 1976

Title: Total Loss of 115 KV Station Service Power Occurs at Connecticut Yankee

The failure sequence was:

1. During refueling, maintenance was being performed on one of the two incoming 115 KV station service lines.
2. While switching to restore this line to service, protective relays sensed a spurious 115 KV fault signal (due to improperly located backup relaying potential sensing circuits) and tripped the automatic circuit breaker on the station service transformer associated with the operating 115 KV line, resulting in a loss of offsite power (3 times).
3. The diesel generators started and loaded safety related loads.

Corrective action:

1. The position of the applicable backup relaying potential sensing circuits were changed.

Design purpose of failed system or component:

Offsite power provides power to safety-related plant loads when the unit generator is not operating.

Unavailability of system per WASH 1400:* Offsite power: 2×10^{-5}/hr.

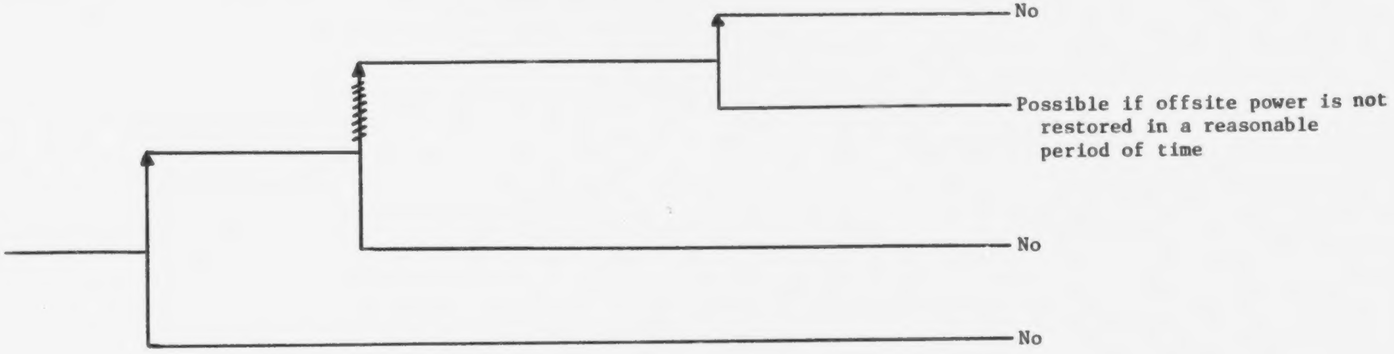
Unavailability of component per WASH 1400:* --

---

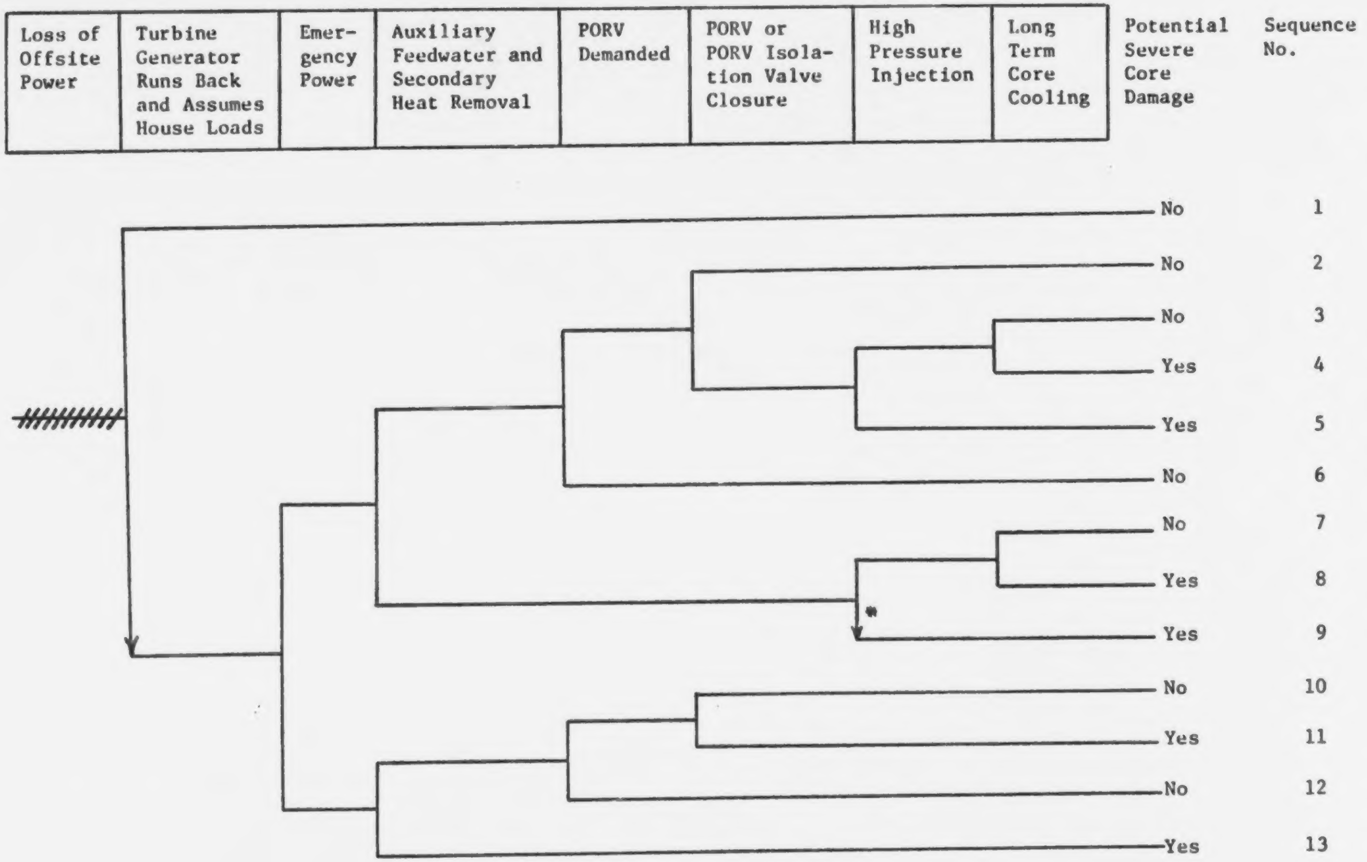
*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

293

Reactor in refueling shutdown	Maintenance in progress on one of two incoming 115kv station service lines	During switching to restore line to service, improperly placed backup relaying potential sensing circuits sense a spurious 115kv fault and trip a circuit breaker associated with the operating 115kv line	Diesel generators start and assume safety related loads	Potential Severe Core Damage


No
Possible if offsite power is not restored in a reasonable period of time
No
No

NSIC 115875 – Actual Occurrence for Total Loss of 115kv Station Power Occurs at Connecticut Yankee


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
*
Yes 9
No 10
Yes 11
No 12
Yes 13

NSIC 115875 — Sequence of Interest for Total Loss of 115kv Station Power Occurs at Connecticut Yankee

*not part of mitigation procedure

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 115875

DATE OF LER: July 9, 1976

DATE OF EVENT: June 26, 1976

SYSTEM INVOLVED: offsite power

COMPONENT INVOLVED: backup relaying potential sensing circuits

CAUSE: improper placement of sensing circuits

SEQUENCE OF INTEREST: loss of offsite power

ACTUAL OCCURRENCE: loss of offsite power during refueling.

REACTOR NAME: Haddam Neck

DOCKET NUMBER: 50-213

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 575 MWe

REACTOR AGE: 9 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Stone & Webster

OPERATORS: Connecticut Yankee Atomic Power Co.

LOCATION: 13 mi. East of Meridan, Conn.

DURATION: N/A

PLANT OPERATING CONDITION: refueling

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ______________________

DISCOVERY METHOD: operational event

COMMENT:

296

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: **116212**

Date: **August 3, 1976**

Title: **Inadvertant Load Shedding on Safety-Related Buses at Millstone 2**

The failure sequence was:

1. On July 5, 1976, Millstone 2 was decreasing load to repair a leaking feedwater heater when a reactor trip occurred due to a steam generator transient. The station loads had already been transferred to an offsite source. As a result of the trip and the fact that the transmission network was very lightly loaded and lightly supplied at the time, the 345 KV grid voltage dropped to 335 KV (its normal voltage is approximately 360 KV) and remained there for over one hour.

2. Following the trip, certain non-safety related 480 V equipment did not start due to blown control power fuses in the equipment motor control centers. It was concluded that the fuses blew because of the low line voltage, perhaps compounded by inherent cable voltage drops to the individual motor control centers.

3. The utility concluded that under similar low voltage conditions 480 V safety-related equipment might not operate and tests were made to determine the minimum voltage (410 V) at which equipment operability could be assured. cont.

Corrective action:

1. The undervoltage trip logic was modified (1) to prevent additional load shedding once the diesels have started, and (2) to provide dual undervoltage trip setpoints which would allow for short-term grid transients.

2. The transformer taps were changed to optimize in-plant voltage.

Design purpose of failed system or component:

The emergency power system provides power to safety-related equipment in the event of a loss of offsite power.

Unavailability of system per WASH 1400:* **emergency power: 1 x 10^{-2}/D**

Unavailability of component per WASH 1400:* **--**

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

PRECURSOR DESCRIPTION AND DATA NSIC Accession Number: 116212

failure sequence continued

4. To prevent such a low-voltage occurrence, the ESFAS undervoltage relays were raised to assure the plant would be separated from the grid before grid voltage fell to a point at which equipment operability could not be assured.

5. On July 16, 1976, with the plant at 100% power, a 1500 HP circulating water pump was started. The current inrush from the pump start resulted in a voltage drop on the safety-related bus to below the revised undervoltage trip point.

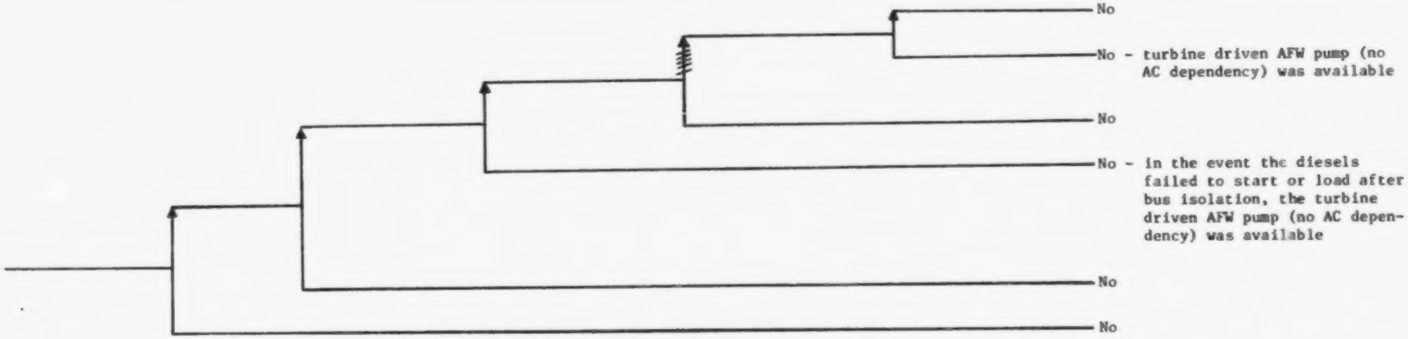
6. The ESAS sensed the undervoltage condition, isolated the safety-related buses from the plant electrical system, tripped the reactor, shed bus loads, started the diesel generators, and connected them to the safety-related buses.

7. When the service water and reactor building closed cooling water pumps were sequenced on to the safety-related buses, the inrush current for each component caused the bus voltage to drop below the new undervoltage setpoint for each component and these loads tripped off the buses.

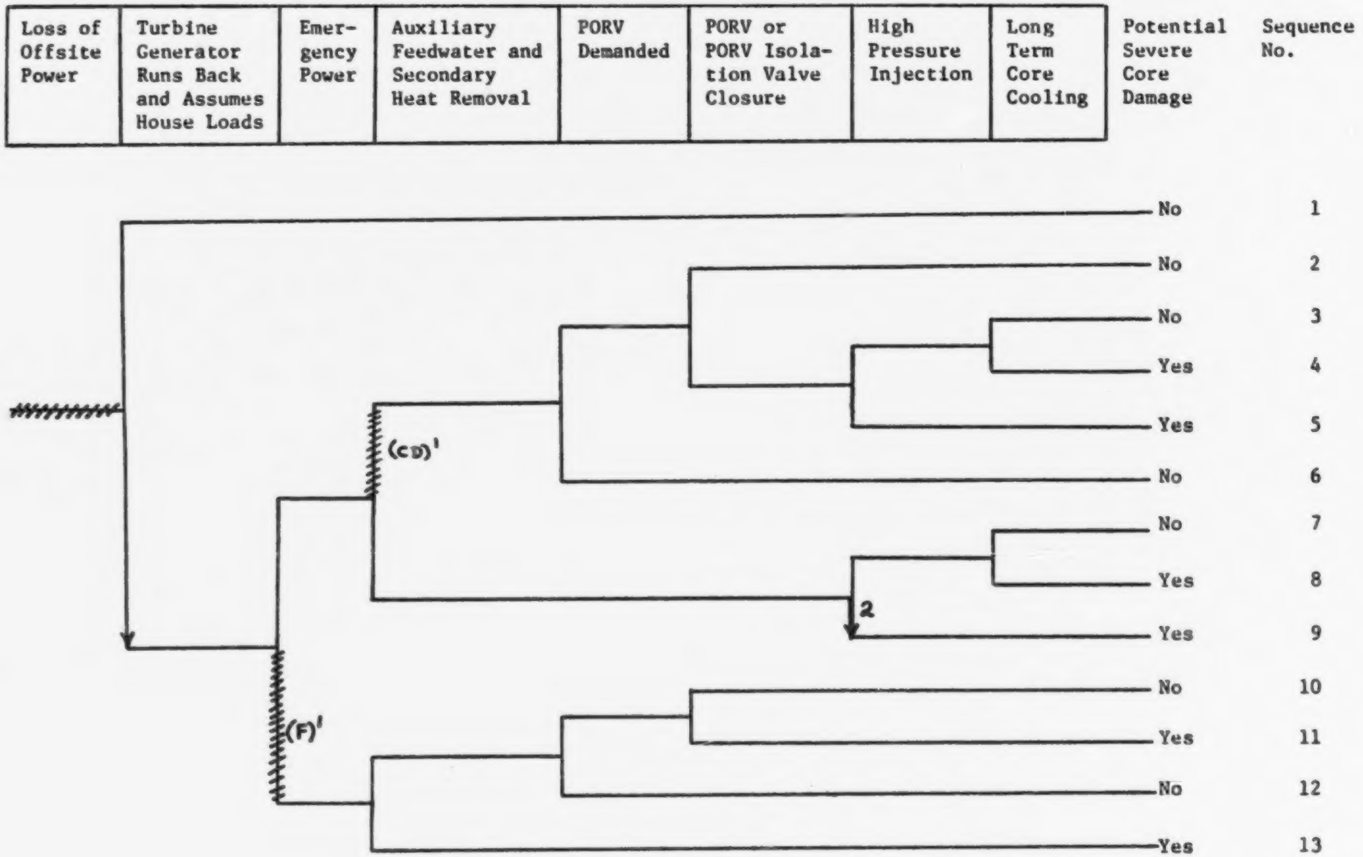
8. At the completion of the load sequence the emergency buses were energized but the main loads were unconnected due to the additional load shed signals.

9. The undervoltage bistables were immediately reset to their previous setpoints and the reserve station transformer connected to the safety-related buses. The service water and reactor building closed cooling water pumps were started satisfactorily five minutes after their previous trip.

Reactor at 100% power and under-voltage trip set-points reset to assure adequate voltage to safety-related components in the evnet of degraded grid voltage	Operator starts circulating water pump	Pump current in-rush results in voltage drop on safety-related bus below revised undervoltage trip point	Safety-related buses isolate, reactor trip, bus load shed, diesel generators start and connect to safety-related buses	Service water and reactor building closed cooling water pumps trip on undervoltage during sequencing due to current in-rush dropping bus voltage below revised undervoltage trip setpoints	Operator resets undervoltage trip setpoints and successfully loads service water and reactor building closed cooling water pumps	Potential Severe Core Damage


No
No - turbine driven AFW pump (no AC dependency) was available
No
No - in the event the diesels failed to start or load after bus isolation, the turbine driven AFW pump (no AC dependency) was available
No
No

NSIC 116212 – Actual Occurrence for Inadvertent Load Shedding on Safety-Related Buses at Millstone 2


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
Yes 9
No 10
Yes 11
No 12
Yes 13
(CD)'
(F)'
2

NSIC 116212 – Sequence of Interest for Inadvertent Load Shedding on Safety-Related Buses at Millstone 2

[1] Success requies immediate operator action to reset undervoltage trip setpoints and reload safety-related loads.

[2] Operating procedures did not provide for use of HPI for core cooling

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: **116212**

DATE OF LER: **August 3, 1976**

DATE OF EVENT: **July 20, 1976**

SYSTEM INVOLVED: **emergency power**

COMPONENT INVOLVED: **undervoltage trip setpoints**

CAUSE: **Undervoltage trip setpoints incorrectly set, human error**

SEQUENCE OF INTEREST: **loss of offsite power**

ACTUAL OCCURRENCE: **apparent loss of offsite power and failure of major safety related components to load.**

REACTOR NAME: **Millstone 2**

DOCKET NUMBER: **50-336**

REACTOR TYPE: **PWR**

DESIGN ELECTRICAL RATING: **870** MWe

REACTOR AGE: **.8** yr

VENDOR: **Combustion Engineering**

ARCHITECT-ENGINEERS: **Bechtel**

OPERATORS: **Northeast Nuclear Energy Company**

LOCATION: **5 miles SW of New London, Conn.**

DURATION: **N/A**

PLANT OPERATING CONDITION: **100% power**

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________________

DISCOVERY METHOD: **operational event**

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 116780

Date: August 10, 1976

Title: Gas Turbine Becomes Inoperable During Plant Trip at Millstone - 1

The failure sequence was:

1. Salt buildup on the 345 kv lines and insulators caused a loss of offsite power.
2. The gas turbine and diesel generator started, however, the gas turbine tripped. It was restarted but again tripped after about 8 minutes. Investigation revealed that the trips were due to improper alignment of the gas turbine generators auxiliaries.

Corrective action:

The gas turbine auxiliaries were aligned.

Design purpose of failed system or component:

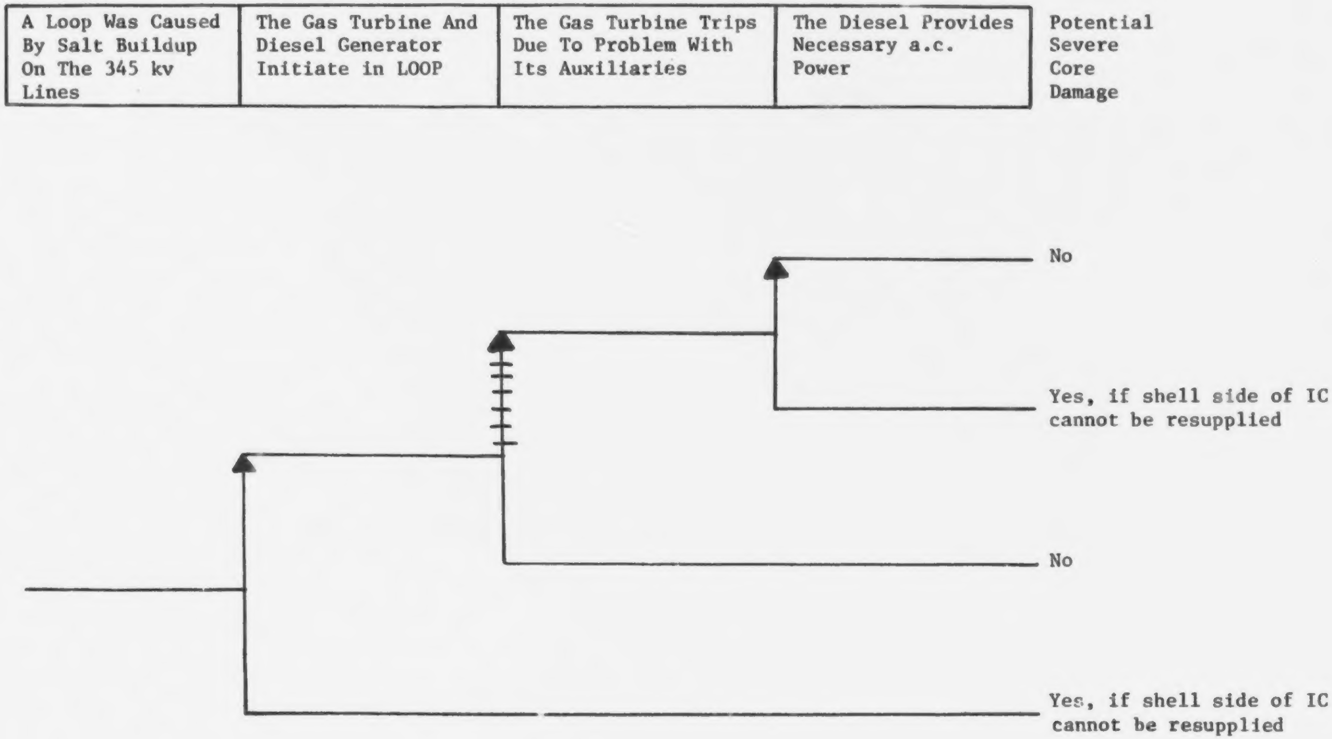
Emergency power provides (a.c.) electric power when offsite power sources are unavailable.

Unavailability of system per WASH 1400:* Emergency power: 10^{-3}/D

Unavailability of component per WASH 1400:*

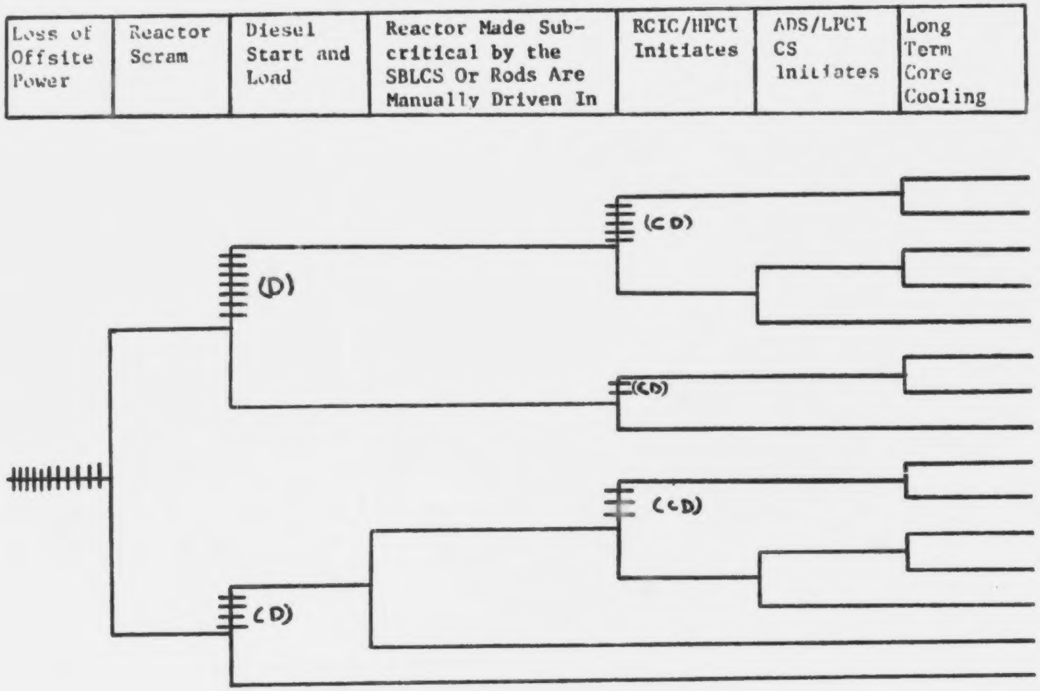
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


A Loop Was Caused By Salt Buildup On The 345 kv Lines
The Gas Turbine And Diesel Generator Initiate in LOOP
The Gas Turbine Trips Due To Problem With Its Auxiliaries
The Diesel Provides Necessary a.c. Power
Potential Severe Core Damage
No
Yes, if shell side of IC cannot be resupplied
No
Yes, if shell side of IC cannot be resupplied

NSIC 116780 — Actual Occurrence of Gas Turbine Becomes Inoperable During Plant Trip at Millstone 1

BEST DOCUMENT AVAILABLE


Loss of Offsite Power
Reactor Scram
Diesel Start and Load
Reactor Made Subcritical by the SBLCS Or Rods Are Manually Driven In
RCIC/HPCI Initiates
ADS/LPCI CS Initiates
Long Term Core Cooling
(D)
(CD)
(CD)
(CD)
(D)

Potential Severe Core Damage	Sequence No.
No	1
Yes	2
No	3
Yes	4
Yes	5
No	6
Yes	7
Yes	8
No	9
Yes	10
No	11
Yes	12
Yes	13
Yes	14
Yes	15

NSIC 116780 — Sequence of Interest of Gas Turbine Becomes Unavailable at Millstone 1

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 116780

DATE OF LER: August 24, 1976

DATE OF EVENT: August 10, 1976

SYSTEM INVOLVED: Electric Power

COMPONENT INVOLVED: Gas turbine, Transmission line.

CAUSE: Weather, maintenance

SEQUENCE OF INTEREST:

ACTUAL OCCURRENCE: Gas Turbine Becomes Inoperable During Plant Trip.

REACTOR NAME: Millstone 1

DOCKET NUMBER: 245

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 660 MWe

REACTOR AGE: 5.9 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: Ebasco

OPERATORS: North East Nuclear Energy Company

LOCATION: Five miles SW of New London

DURATION: N/A

PLANT OPERATING CONDITION: Plant had just tripped.

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c) made inoperable]; (d) ____________

DISCOVERY METHOD: Operational transient

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 117944

Date: September 3, 1976

Title: Offsite Power Lost at Millstone 2

The failure sequence was:

1. During a hurricane, severe salt contamination caused several switchyard failures, resulting in a loss of offsite power and a reactor trip from 100% power.
2. The diesel generators, which had been running prior to the trip, provided power to safety related loads.

Corrective action:

Switchyard equipment and insulators were washed down, damaged circuit breakers were repaired, and the offsite transmission lines returned to service.

Design purpose of failed system or component:

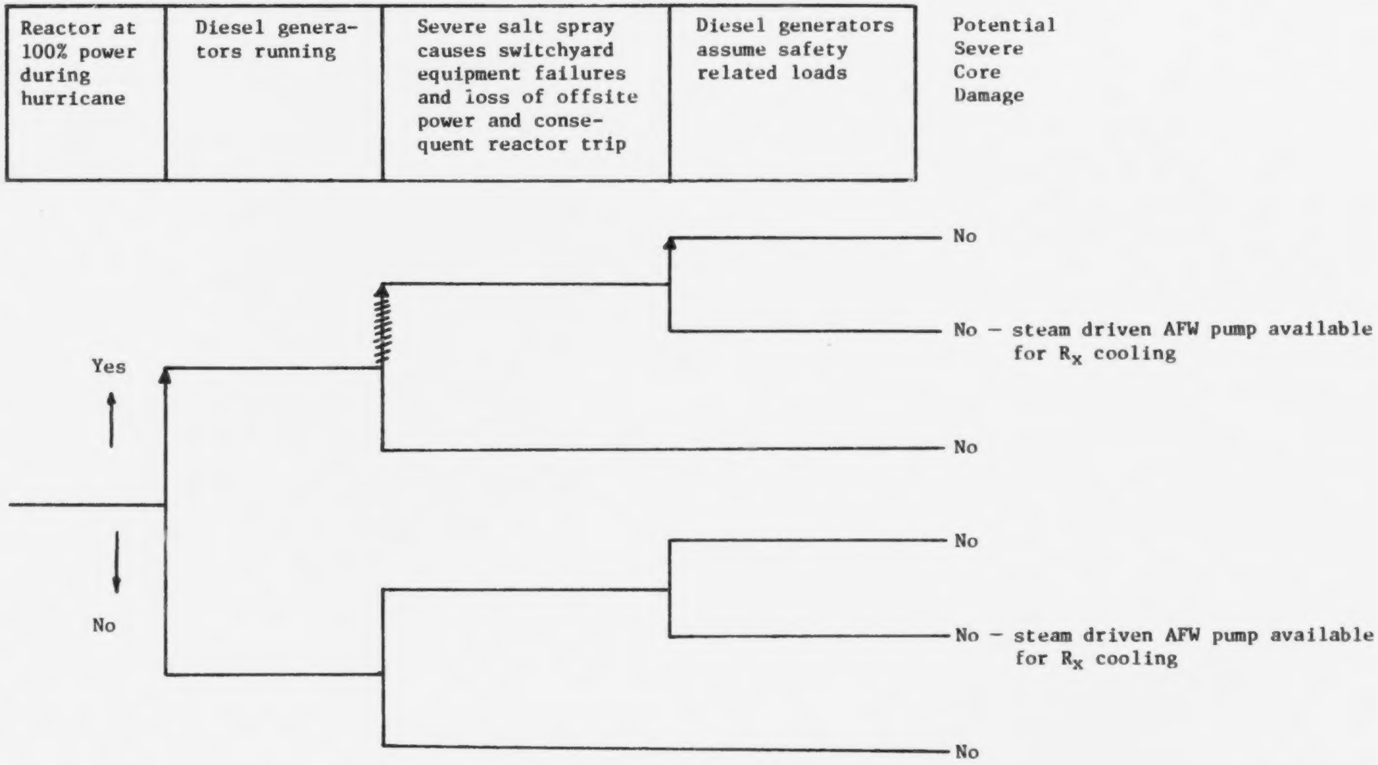
Offsite power provides the preferred source of power to safety related loads when the unit generator is not available.

Unavailability of system per WASH 1400:* off-site power: 2×10^{-5}/hr

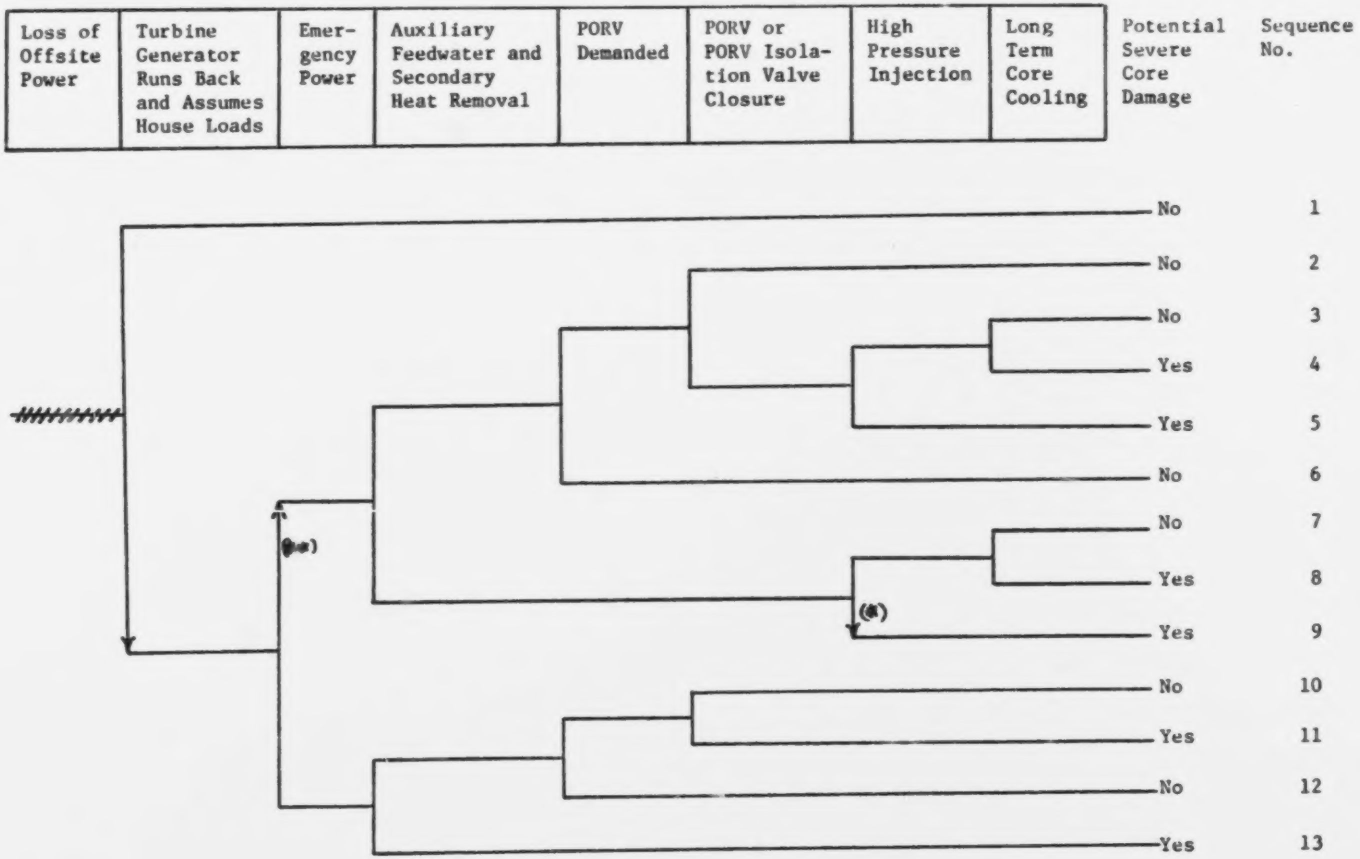
Unavailability of component per WASH 1400:* —

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor at 100% power during hurricane
Diesel generators running
Severe salt spray causes switchyard equipment failures and loss of offsite power and consequent reactor trip
Diesel generators assume safety related loads
Potential Severe Core Damage
Yes
No
No
No – steam driven AFW pump available for R_x cooling
No
No
No – steam driven AFW pump available for R_x cooling
No

NSIC 117944 – Actual Occurrence for Offsite Power Lost at Millstone 2


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
(**)
(*)
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
Yes 9
No 10
Yes 11
No 12
Yes 13

NSIC 117944 — Sequence of Interest for Offsite Power Lost at Millstone 2

*Use of HPI after AFW failure not included in mitigation procedures.

**diesel generators running but not loaded

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 117944

DATE OF LER: September 3, 1976

DATE OF EVENT: August 10, 1976

SYSTEM INVOLVED: offsite power

COMPONENT INVOLVED: switchyard insulators and equipment

CAUSE: severe salt spray wetting during hurricane

SEQUENCE OF INTEREST: loss of offsite power

ACTUAL OCCURRENCE: reactor trip with loss of offsite power and diesel generators previously started and running

REACTOR NAME: Millstone 2

DOCKET NUMBER: 50-336

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 830 MWe

REACTOR AGE: .8 yr

VENDOR: Combustion Engineering

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Northeast Nuclear Energy Co.

LOCATION: 5 miles SW of New London, Conn.

DURATION: N/A

PLANT OPERATING CONDITION: 100% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________

DISCOVERY METHOD: during operation

COMMENT: –

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 120207

Date: November 12, 1976

Title: Failure of Charging Pump Supply Valves to Open from RWST at Trojan

The failure sequence was:

With plant in shutdown, chemicals were added to S/G too fast and a SI signal actuated. The charging pump suction valves from the RWST failed to open. Both valves had been closed manually too hard to prevent leakage. Torque switch cut out before valve was off seat.

Corrective action:

Torque switch bypass switch had drifted so that torque switch cut in before valve was off the seat. Bypass switch was adjusted.

Design purpose of failed system or component:

The system provides for injection of high-concentration boric acid to ensure subcriticality following a steam line break.

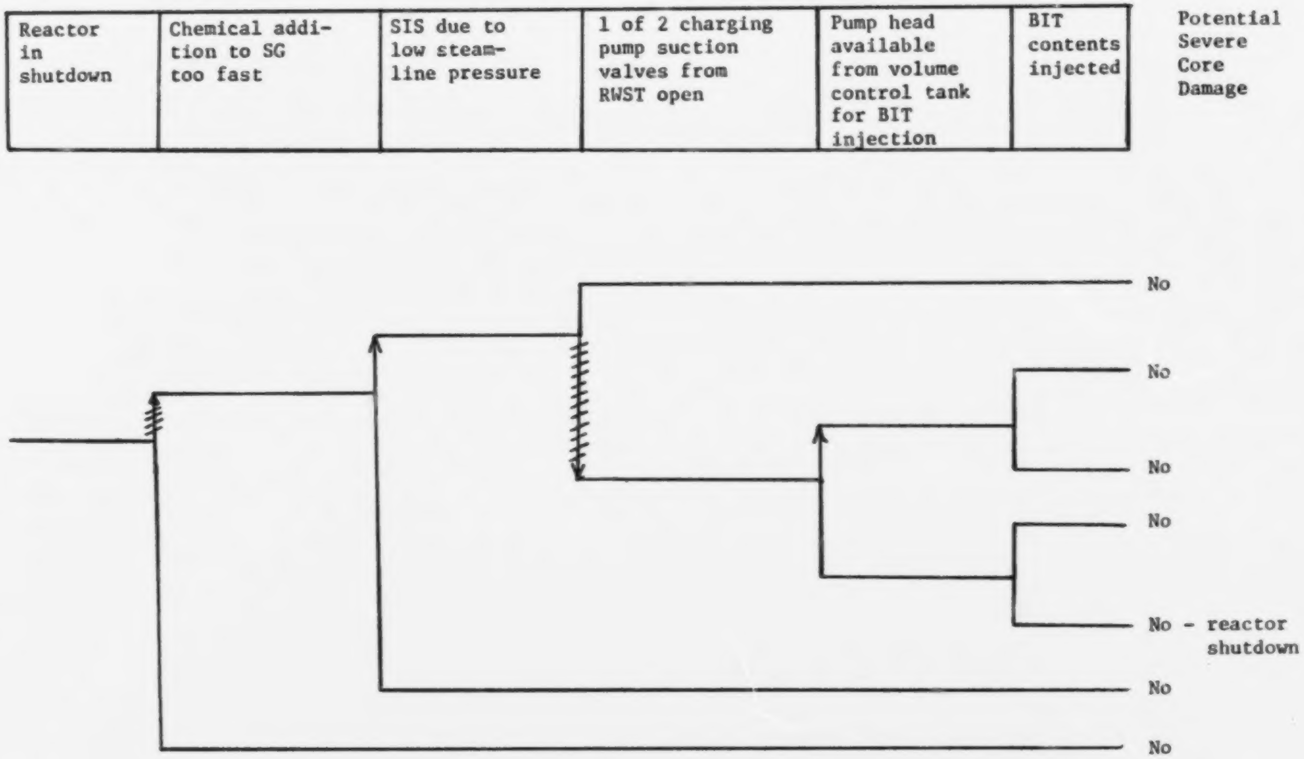
RWST provides pump head for BIT injection.

Unavailability of system per WASH 1400:* —

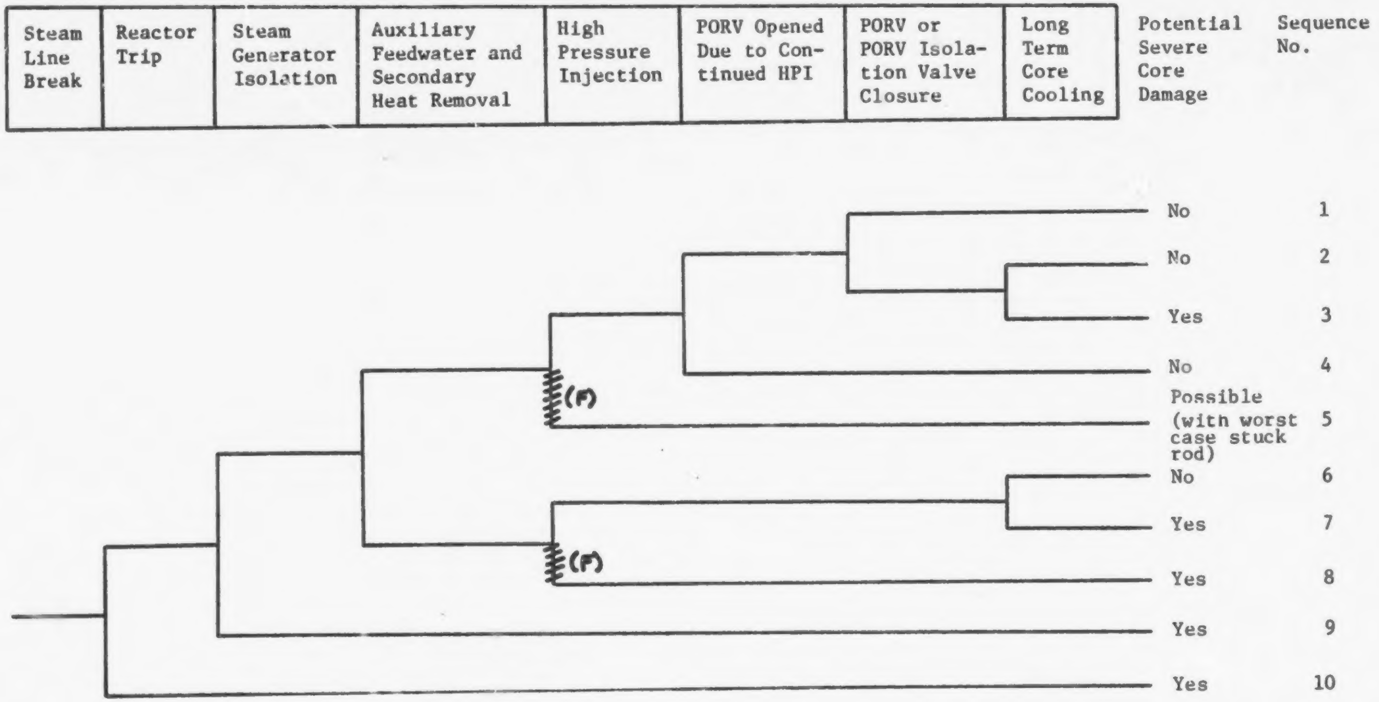
Unavailability of component per WASH 1400:* motor operated valve: 1×10^{-3}/D

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor in shutdown
Chemical addition to SG too fast
SIS due to low steamline pressure
1 of 2 charging pump suction valves from RWST open
Pump head available from volume control tank for BIT injection
BIT contents injected
Potential Severe Core Damage
No
No
No
No
No - reactor shutdown
No
No

NSIC 120207 — Actual Occurrence for Failure to Open of Both Charge Pump Supply Valves from RWST at Trojan


Steam Line Break
Reactor Trip
Steam Generator Isolation
Auxiliary Feedwater and Secondary Heat Removal
High Pressure Injection
PORV Opened Due to Continued HPI
PORV or PORV Isolation Valve Closure
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
(F)
(F)
No 1
No 2
Yes 3
No 4
Possible (with worst case stuck rod) 5
No 6
Yes 7
Yes 8
Yes 9
Yes 10

NSIC 120207 — Sequence of Interest for Failure of Both Charging Pump Suction Valves from RWST to Open at Trojan

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 120207

DATE OF LER: November 12, 1976

DATE OF EVENT: October 28, 1976

SYSTEM INVOLVED: HPI

COMPONENT INVOLVED: Valves

CAUSE: Switch set point drift, human error

SEQUENCE OF INTEREST: Steam line break.

ACTUAL OCCURRENCE: Chemical addition to steam generator too fast.

REACTOR NAME: Trojan

DOCKET NUMBER: 50-344

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 1130 MWe

REACTOR AGE: 0.9 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Portland General Electric Co.

LOCATION: 42 miles north of Portland, OR

DURATION: 360(a) hours

PLANT OPERATING CONDITION: In shutdown

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ________________

DISCOVERY METHOD: Operational event

COMMENT: –

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 120293

Date: November 11, 1976

Title: Plant Service Water Strainers Plug at Hatch 1

The failure sequence was:

1. While operating at 85% power, both division I and II service water strainers plugged, thus reducing the plant service water to the turbine and Reactor building. The strainer drive motors had failed.

Corrective action:

The strainers were manually back flushed twice per shift until the drive motors were repaired.

Design purpose of failed system or component:

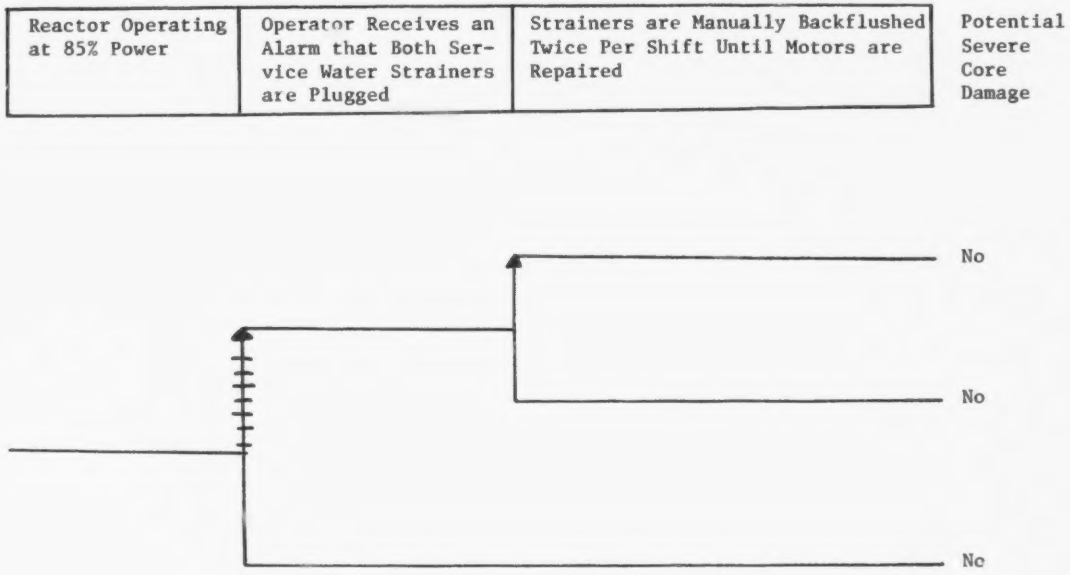
The drive motors power the pumps that keep the service water strainers clean.

Unavailability of system per WASH 1400:* High Pressure Service Water: 1.1×10^{-4}

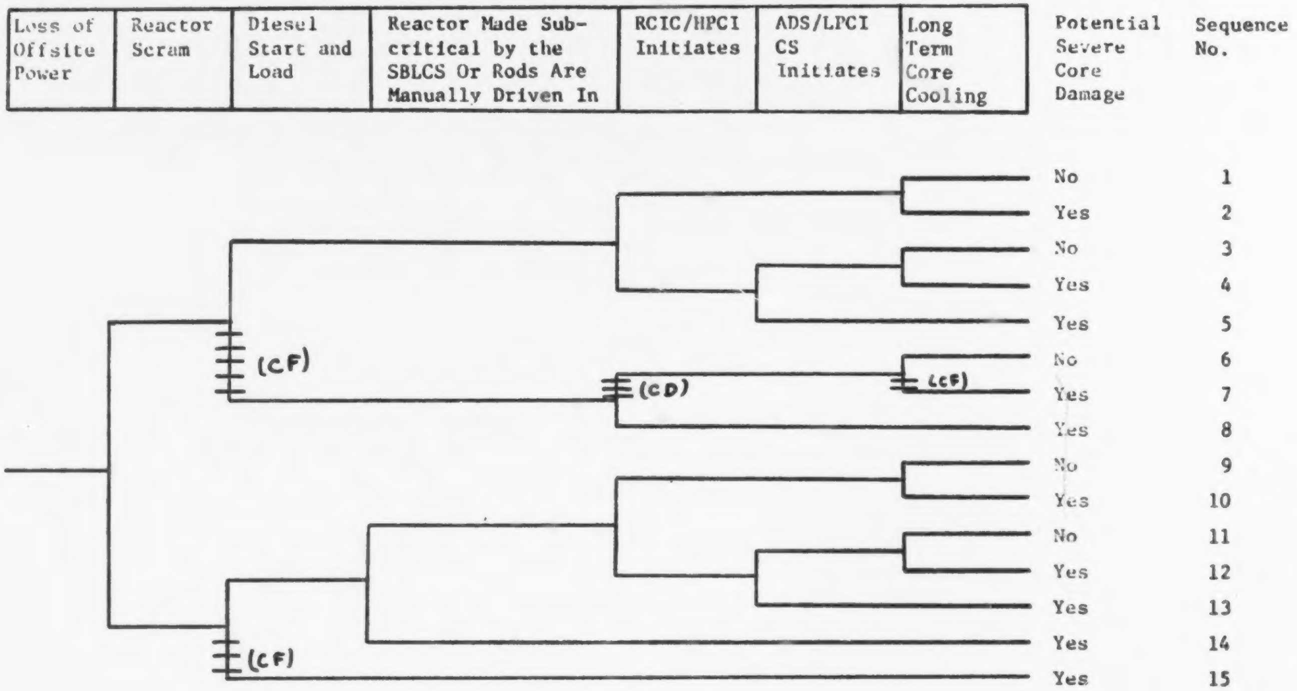
Unavailability of component per WASH 1400:* Pumps failure to run: 3×10^{-5}/Hr

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor Operating at 85% Power
Operator Receives an Alarm that Both Service Water Strainers are Plugged
Strainers are Manually Backflushed Twice Per Shift Until Motors are Repaired
Potential Severe Core Damage
No
No
No

NSIC 120293 — Actual Occurrence of Plant Service Water Strainers Plug at Hatch 1


Loss of Offsite Power
Reactor Scram
Diesel Start and Load
Reactor Made Sub-critical by the SBLCS Or Rods Are Manually Driven In
RCIC/HPCI Initiates
ADS/LPCI CS Initiates
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
Yes 2
No 3
Yes 4
Yes 5
No 6
Yes 7
Yes 8
No 9
Yes 10
No 11
Yes 12
Yes 13
Yes 14
Yes 15
(CF)
(CD)
(CF)
(CF)

NSIC 120293 — Sequence of Interest of Plant Service Water Strainers Plug at Hatch 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 120293

DATE OF LER: November 11, 1970

DATE OF EVENT: Unknown

SYSTEM INVOLVED: Service Water

COMPONENT INVOLVED: Strainers

CAUSE: Plugging

SEQUENCE OF INTEREST: Loop

ACTUAL OCCURRENCE: Plant Service Water Strainers Plug at Hatch 1

REACTOR NAME: Hatch 1

DOCKET NUMBER: 50-321

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 777 MWe

REACTOR AGE: 2.2 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: Southern Services

OPERATORS: Georgia Power

LOCATION: Eleven miles N of Baxley, GA

DURATION: 12 (a) hours

PLANT OPERATING CONDITION: 85%

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ______________________

DISCOVERY METHOD:

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: **120439**

Date: **November 1, 1976**

Title: **Two Electromatic Relief Valves Fail at Quad Cities 1**

The failure sequence was:

1. **Failure of valve to open during unit 2 startup test warranted surveillance testing of unit 1 electromatic relief valves.**
2. **Valves 1-203-3E and 1-203-3C were found inoperable, thus rendering the ADS system inoperable (2 out of 5).**
3. **The reactor was shutdown.**
4. **The apparent cause of the valve failure was excessive steam leakage that damaged the pilot valves. Because of the additional steam the pilot valve could not vent below the valve disc to allow the valve to open.**

Corrective action:

Valve 1-203-3E was replaced with a previously overhauled spare.

Valve 1-203-3C was overhauled and reinstalled.

Design purpose of failed system or component:

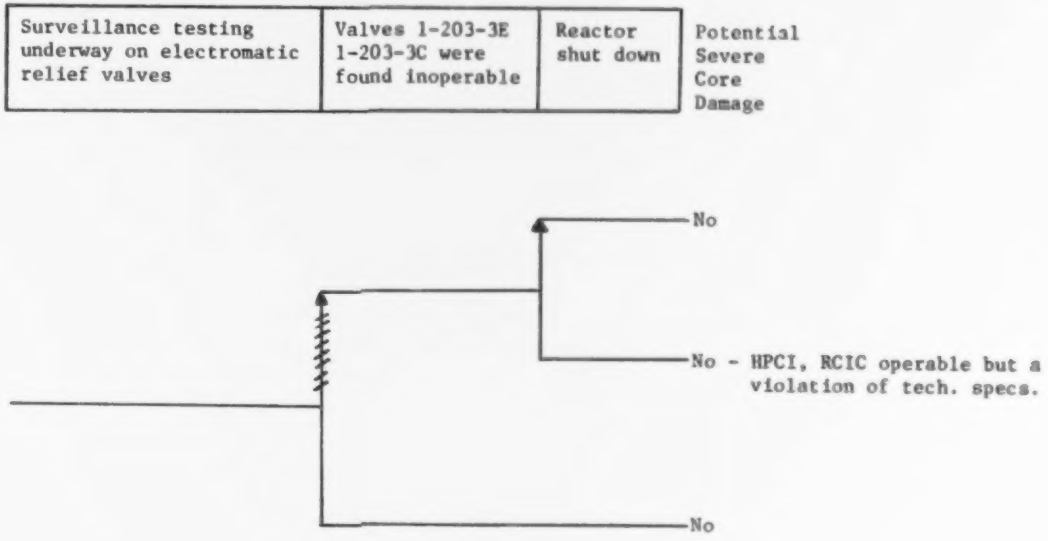
1. **The ADS is designed to lower the system pressure should the high pressure ECCS fail so the low pressure systems can be used to provide cooling.**
2. **The electromatic relief valves allow for a controlled depressurization of the reactor.**

Unavailability of system per WASH 1400:* **ADS: 5×10^{-3}/D**

Unavailability of component per WASH 1400:* **relief valve, failure to open: 1×10^{-5}/D**

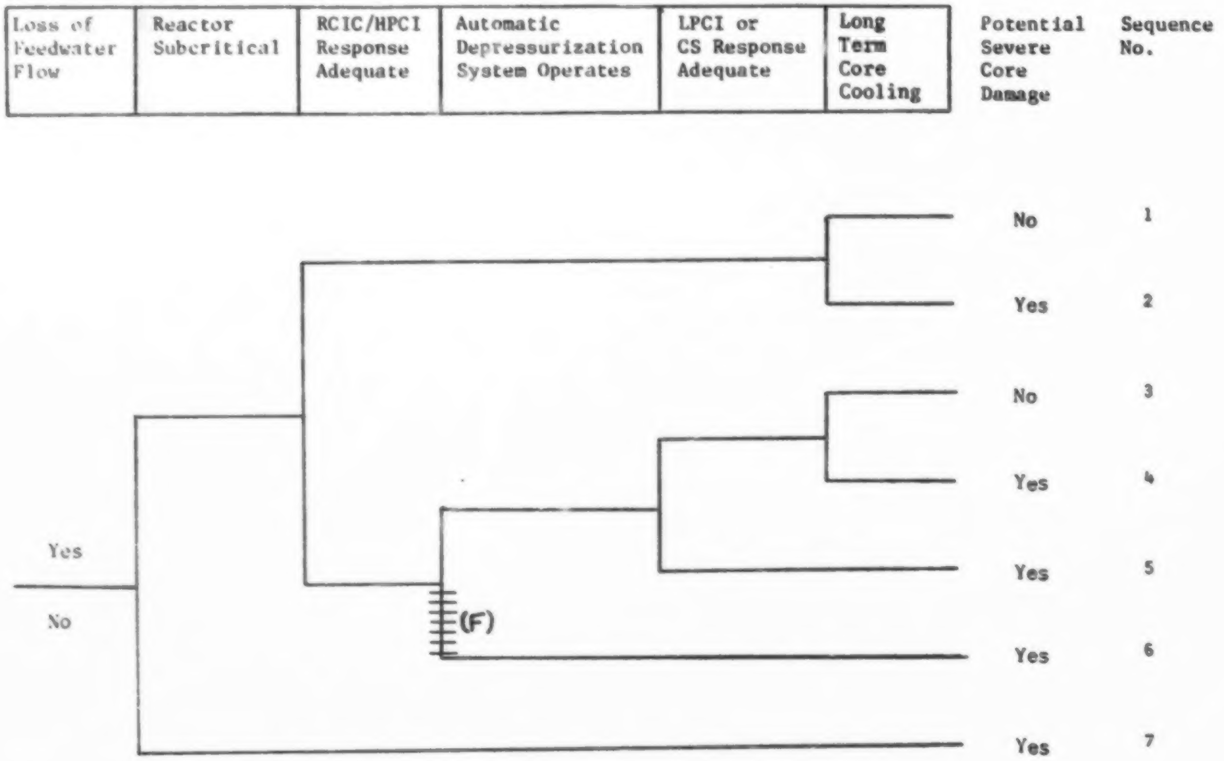
---

[*] Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Surveillance testing underway on electromatic relief valves
Valves 1-203-3E 1-203-3C were found inoperable
Reactor shut down
Potential Severe Core Damage
No
No - HPCI, RCIC operable but a violation of tech. specs.
No

NSIC 120439 — Actual Occurrence of Two Electromatic Relief Valves Fail at Quad Cities 1

BEST DOCUMENT AVAILABLE


Loss of Feedwater Flow
Reactor Subcritical
RCIC/HPCI Response Adequate
Automatic Depressurization System Operates
LPCI or CS Response Adequate
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
(F)
No 1
Yes 2
No 3
Yes 4
Yes 5
Yes 6
Yes 7

NSIC 120439 - Sequence of Interest for Two Electromatic Relief Valves Fail at Quad Cities 1

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 120439

DATE OF LER: November 12, 1976

DATE OF EVENT: November 1, 1976

SYSTEM INVOLVED: Automatic Depressurization System

COMPONENT INVOLVED: relief valve

CAUSE: failure of the valve disc due to excessive steam leakage

SEQUENCE OF INTEREST: loss of feedwater

ACTUAL OCCURRENCE: two electromatic relief valves fail at Quad Cities 1

REACTOR NAME: Quad Cities 1

DOCKET NUMBER: 50-254

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 789 MWe

REACTOR AGE: 4.9 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: Sargent & Lundy

OPERATORS: Commonwealth Edison

LOCATION: 20 miles NE of Moline, Ill.

DURATION: 1290 (a) hours

PLANT OPERATING CONDITION: operating but power unknown

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ______________________

DISCOVERY METHOD: surveillance testing

COMMENT: Valves were target rock relief valves. See also 120443.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: **120443**

Date: **November 1, 1976**

Title: **Electromatic Relief Valves Fail at Quad Cities 2**

The failure sequence was:

1. **During testing of valves prior to startup, two out of five electromatic relief valves (2-203-3C and 2-203-3E) failed to open. Operation of fewer than 4/5 relief valves results in ADS failure.**
2. **The reactor startup was terminated.**

Corrective action:

1. **The valves were removed, disassembled and rebuilt. Valve 2-203-3C was reinstalled while valve 2-203-3E was replaced with a previously rebuilt spare.**

Design purpose of failed system or component:

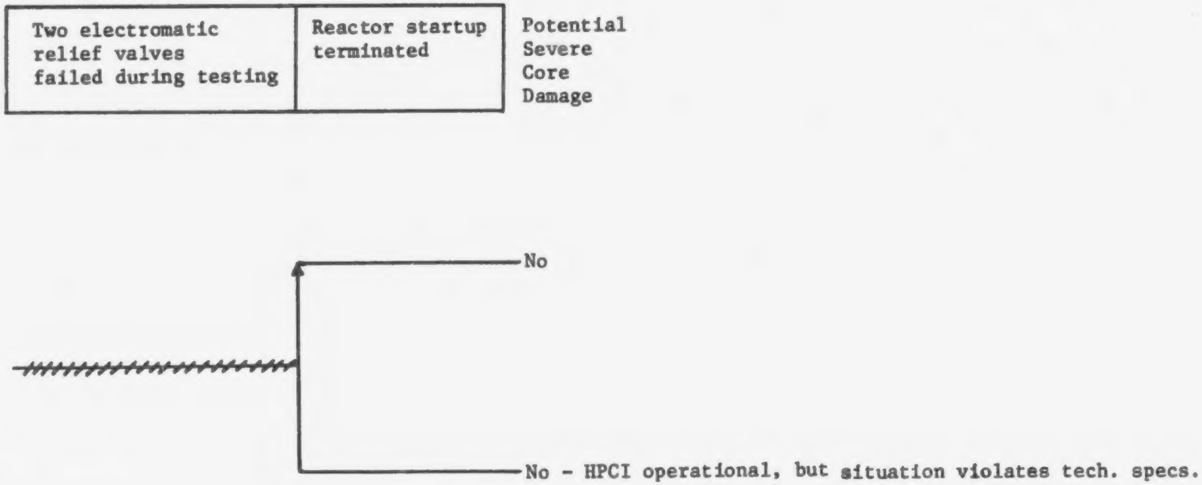
1. **The ADS is designed to lower system pressure should the high pressure ECCS fail. Once the pressure has been adequately reduced the low pressure ECCS can provide core cooling.**
2. **The electromatic relief valves allow for a controlled depressurization of the system.**

Unavailability of system per WASH 1400:* **ADS: 5×10^{-3}/D**

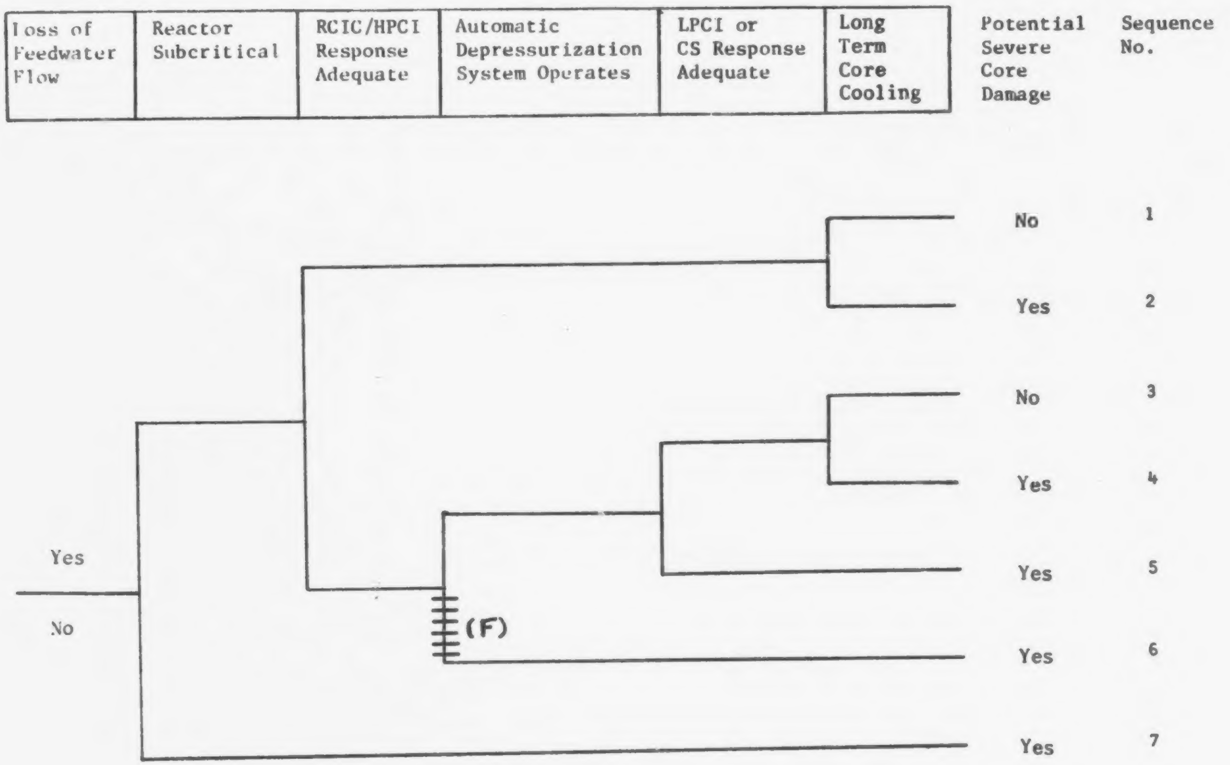
Unavailability of component per WASH 1400:* **relief valve, failure to open: 1×10^{-5}/D**

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Two electromatic relief valves failed during testing
Reactor startup terminated
Potential Severe Core Damage
No
No - HPCI operational, but situation violates tech. specs.

NSIC 120443 - Actual Occurrence for Electromatic Relief Valves Fail at Quad Cities 2


Loss of Feedwater Flow
Reactor Subcritical
RCIC/HPCI Response Adequate
Automatic Depressurization System Operates
LPCI or CS Response Adequate
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
(F)
No 1
Yes 2
No 3
Yes 4
Yes 5
Yes 6
Yes 7

NSIC 120443 - Sequence of Interest for Two Electromatic Relief Valves Fail at Quad Cities 2

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 120443

DATE OF LER: November 15, 1976

DATE OF EVENT: November 1, 1976

SYSTEM INVOLVED: ADS

COMPONENT INVOLVED: electromatic relief valve

CAUSE: mechanical failure of the valve pilots

SEQUENCE OF INTEREST: loss of feedwater

ACTUAL OCCURRENCE: failure of two electromatic relief valves at Quad Cities 2

REACTOR NAME: Quad Cities 2

DOCKET NUMBER: 50-265

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 789 MWe

REACTOR AGE: 4.6 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: Sargent & Lundy

OPERATORS: Commonwealth Edison

LOCATION: 20 miles NE of Moline, Ill.

DURATION: 4380 (a) hours

PLANT OPERATING CONDITION: startup

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________________

DISCOVERY METHOD: startup testing

COMMENT: The same valve and valve location, -203-3C and -203-3E failed in unit one on the same day. See 120439.

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 120995

Date: November 11, 1976

Title: Suction Valves to Decay Heat Pumps Fail to Open at Rancho Seco

The failure sequence was:

1. During testing, neither sump isolation valve opened when an open signal was applied.

Corrective action;

1. The valves were moved manually and then cycled electrically. The bypass switches for the valve torque switches was adjusted to permit more motor turns before the bypass switches dropped out.

Design purpose of failed system or component:

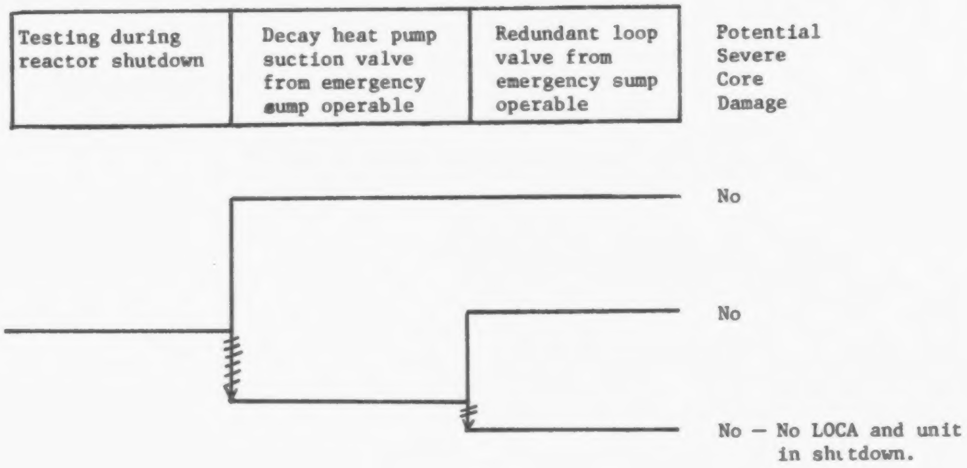
1. The valves provide for fluid flow from the sump to the low pressure and high pressure recirculation systems and the building spray system (while in the recirculation mode) following a LOCA.

Unavailability of system per WASH 1400:* low pressure recirc: 1.3×10^{-2}/D
high pressure recirc: 9×10^{-3}/D

Unavailability of component per WASH 1400:* motor operated valve, failure to operate. 1×10^{-3}/D

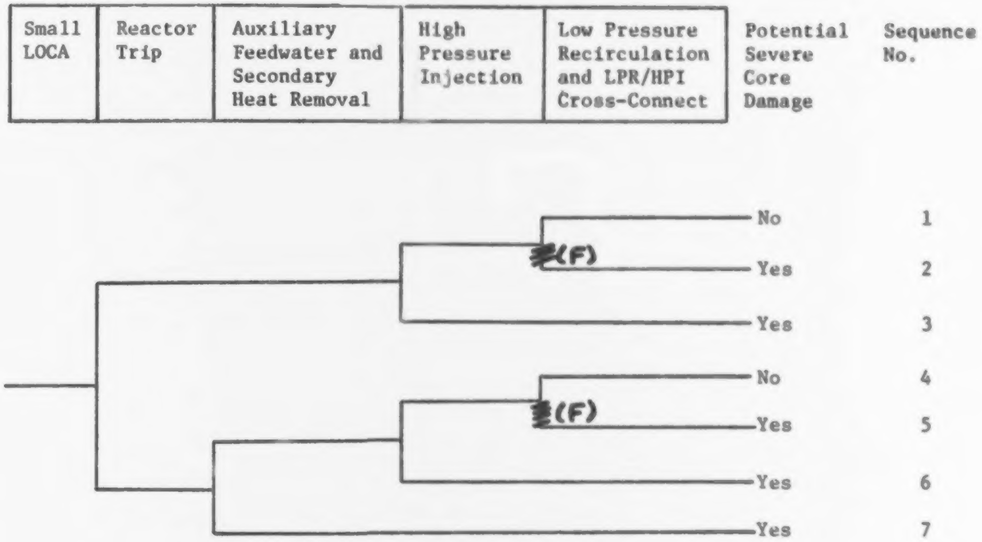
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Testing during reactor shutdown
Decay heat pump suction valve from emergency sump operable
Redundant loop valve from emergency sump operable
Potential Severe Core Damage
No
No
No — No LOCA and unit in shutdown.

NSIC 120995 — Actual Occurrence for Failure of Both Suction Valves to Decay Heat Pumps from Emergency Sump at Rancho Seco

BEST DOCUMENT AVAILABLE


Small LOCA
Reactor Trip
Auxiliary Feedwater and Secondary Heat Removal
High Pressure Injection
Low Pressure Recirculation and LPR/HPI Cross-Connect
Potential Severe Core Damage
Sequence No.
(F)
(F)
No 1
Yes 2
Yes 3
No 4
Yes 5
Yes 6
Yes 7

NSIC 120995 — Sequence of Interest for Both Sump Suction Valves to Decay Heat Pumps Fail to Open at Rancho Seco

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 120995

DATE OF LER: November 22, 1976

DATE OF EVENT: November 11, 1976

SYSTEM INVOLVED: Decay heat

COMPONENT INVOLVED: Pumps

CAUSE: Switch setting incorrect.

SEQUENCE OF INTEREST: LOCA

ACTUAL OCCURRENCE: Valves fail to open for decay heat system.

REACTOR NAME: Rancho Seco

DOCKET NUMBER: 50-312

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 918 MWe

REACTOR AGE: 2.1 yr

VENDOR: B&W

ARCHITECT ENGINEERS: Bechtel

OPERATORS: Sacramento Municipal Utility District

LOCATION: 25 miles SE of Sacramento, CA

DURATION: 360(a) hours

PLANT OPERATING CONDITION: Shut down.

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c) made inoperable]; (d) ________________

DISCOVERY METHOD: During test.

COMMENT: Bypass switches for torque switches were dropping out before valve disc cleared seat resulting in torque switch stopping valve movement.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 121533

Date: January 4, 1977

Title: Two Diesel Generators Trip at Brunswick 1

The failure sequence was:

1. A complete loss of offsite power test was underway with all four diesel generators starting.
2. Diesel generators 3 and 4 soon tripped however for reasons unspecified. The lock out relays were reset and diesel generators 3 and 4 operated properly.
3. About 8 hours later a subsequent test was carried out in which diesels 2, 3, and 4 tripped. The problem was traced to low lube oil pressure and temperature switches.

Corrective action:

1. The setpoint for the lube oil pre-filter heater was increased from 150°F to 230°F.
2. The setpoint for the jacket water system heater was increased from 180°F to 200°F.
3. The setpoint for the low lube oil pressure switch time delay was changed from 30 sec to one minute. (See next page)

Design purpose of failed system or component:

Emergency power provide a.c. power to the plant when offsite sources are unavailable.

Unavailability of system per WASH 1400:* Emergency Power: 10^{-5}/D

Unavailability of component per WASH 1400:* Diesel Engine failure to run: 3×10^{-4}/hr

---

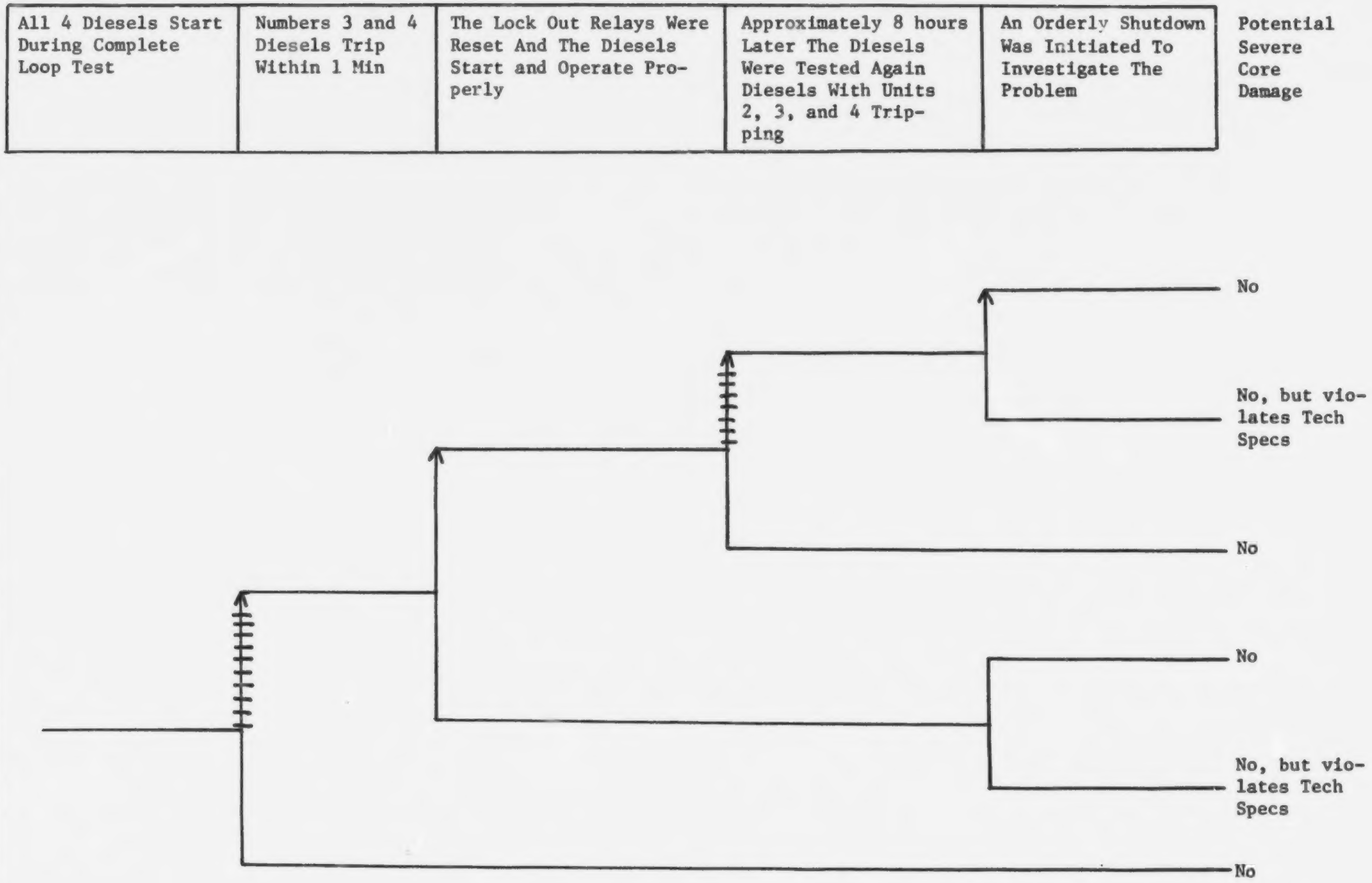
*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

[redacted]

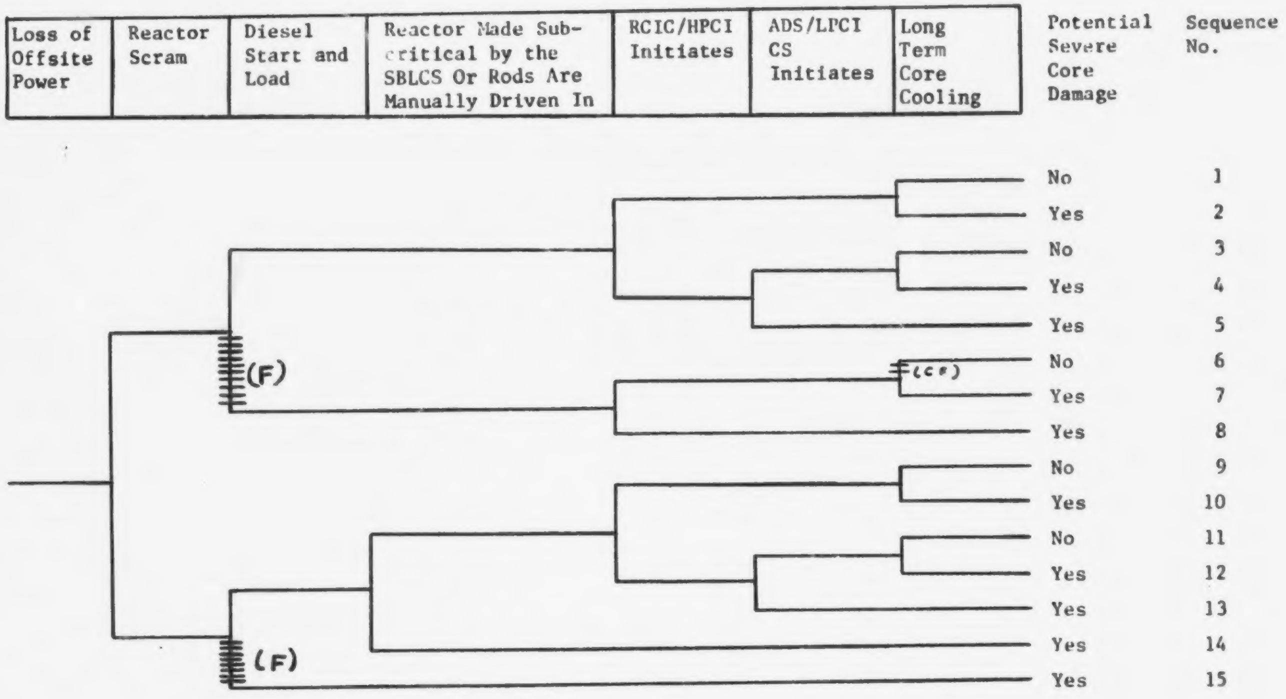
Corrective action: (continued)

4. The diaphragms in the pre-lube/pre-filter valves were found to be broken on units 3 and 4. This prevented heated lube oil from being circulated to the diesel lube oil sump.

BEST DOCUMENT AVAILABLE


All 4 Diesels Start During Complete Loop Test
Numbers 3 and 4 Diesels Trip Within 1 Min
The Lock Out Relays Were Reset And The Diesels Start and Operate Properly
Approximately 8 hours Later The Diesels Were Tested Again Diesels With Units 2, 3, and 4 Tripping
An Orderly Shutdown Was Initiated To Investigate The Problem
Potential Severe Core Damage
No
No, but violates Tech Specs
No
No
No, but violates Tech Specs
No

NSIC 121533 — Actual Occurrence of Two Diesel Generators Trip at Brunswick 1


Loss of Offsite Power
Reactor Scram
Diesel Start and Load
Reactor Made Subcritical by the SBLCS Or Rods Are Manually Driven In
RCIC/HPCI Initiates
ADS/LPCI CS Initiates
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
(F)
(CF)
(F)
No 1
Yes 2
No 3
Yes 4
Yes 5
No 6
Yes 7
Yes 8
No 9
Yes 10
No 11
Yes 12
Yes 13
Yes 14
Yes 15

NSIC 121533 — Sequence of Interest for Two Diesels Trip at Brunswick 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 121533

DATE OF LER: January 18, 1977

DATE OF EVENT: January 4, 1977

SYSTEM INVOLVED: Emergency Power

COMPONENT INVOLVED: Diesel Generators

CAUSE: Mechanical failure, design error

SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE: Two Diesel Generators Trip at Brunswick 1

REACTOR NAME: Brunswick 1

DOCKET NUMBER: 50-325

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 821 MWe

REACTOR AGE: 0.2 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: United Engineers & Constructors

OPERATORS: Carolina Power and Light

LOCATION: Three miles N of Southport, NC

DURATION: 360 (a) hrs.

PLANT OPERATING CONDITION: 76% rated power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start;
(c) made inoperable; [✓ (d)] failed to run

DISCOVERY METHOD: testing

COMMENT:

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 122366

Date: February 2, 1977

Title: SLBIC Trouble Alarm Received at Arkansas Nuclear 1

The failure sequence was:

1. SLBIC trouble alarm received in control room.
2. All four SLBIC sensing lines were found frozen and one pressure transmitter had actuated.

Corrective action;

1. The SLBIC was removed from service, the sensing lines were thawed, and the SBLIC was returned to service.
2. Heat tracing was applied to the sensing lines.

Design purpose of failed system or component:

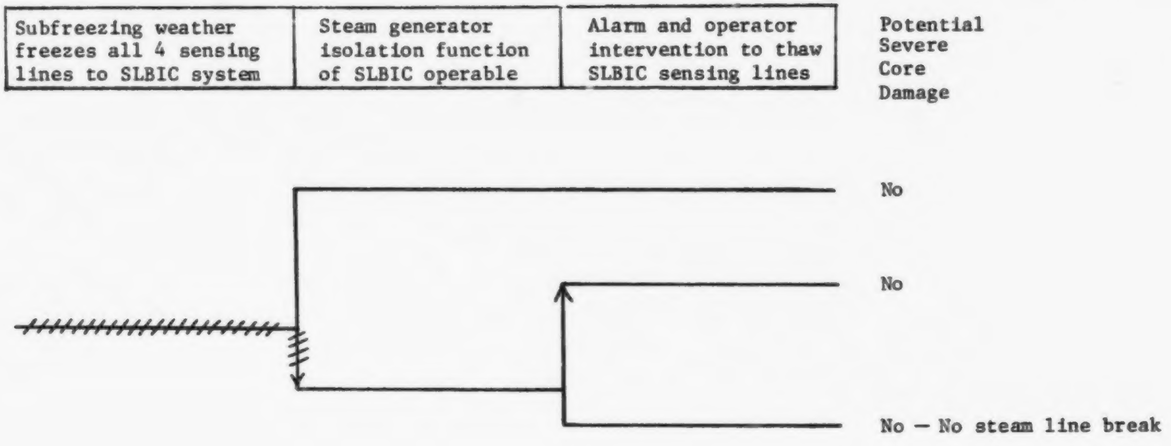
1. Provides signals for steam generator isolation during steam generator break mitigation.

Unavailability of system per WASH 1400:* System not considered in WASH-1400.

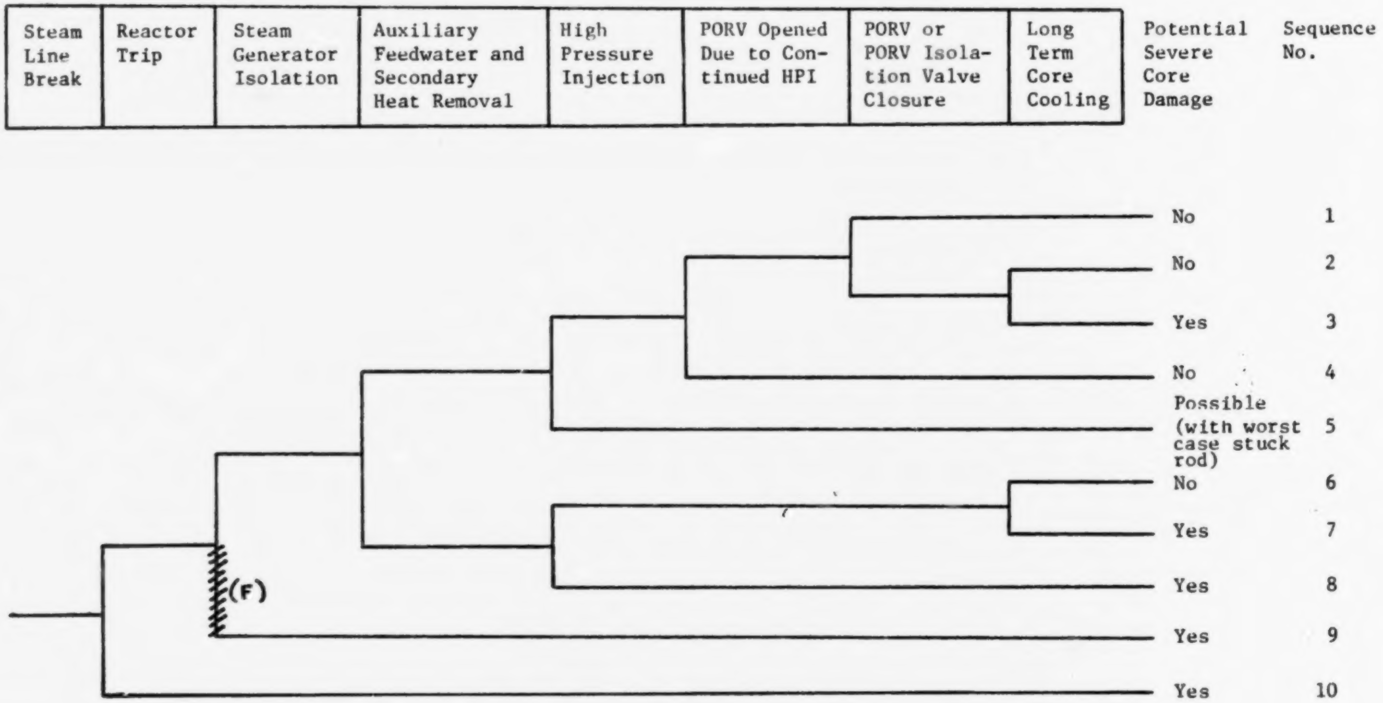
Unavailability of component per WASH 1400:* Specific failure mode not included, general instrumentation failure rate 1×10^{-6}/hr.

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Subfreezing weather freezes all 4 sensing lines to SLBIC system
Steam generator isolation function of SLBIC operable
Alarm and operator intervention to thaw SLBIC sensing lines
Potential Severe Core Damage
No
No
No — No steam line break

NSIC 122366 — Actual Occurrence for Steam Line Break Instrumentation and Control System Failure at Arkansas Nuclear 1


Steam Line Break
Reactor Trip
Steam Generator Isolation
Auxiliary Feedwater and Secondary Heat Removal
High Pressure Injection
PORV Opened Due to Continued HPI
PORV or PORV Isolation Valve Closure
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
(F)
No 1
No 2
Yes 3
No 4
Possible (with worst case stuck rod) 5
No 6
Yes 7
Yes 8
Yes 9
Yes 10

NSIC 122366 — Sequence of Interest for Steam Line Break Instrumentation and Control System Failure at Arkansas Nuclear 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 122366

DATE OF LER: February 2, 1977

DATE OF EVENT: January 10, 1977

SYSTEM INVOLVED: Steam Line Break Instrumentation & Control (SLBIC)

COMPONENT INVOLVED: Sensing lines

CAUSE: Sensing lines frozen

SEQUENCE OF INTEREST: Steam line break.

ACTUAL OCCURRENCE: Sensing lines frozen.

REACTOR NAME: Arkansas Nuclear 1

DOCKET NUMBER: 50-313

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 850 MWe

REACTOR AGE: 2.6 yr

VENDOR: B&W

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Arkansas Power & Light Co.

LOCATION: 6 miles NW of Russelville, Arkansas

DURATION: 8(a) hours

PLANT OPERATING CONDITION: 92% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ________________

DISCOVERY METHOD: Operational event

COMMENT: —

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 123118

Date: March 9, 1977

Title: Both AFW Pumps Unavailable at Rancho Seco

The failure sequence was:

1. With the reactor at 100% power, AFW pump P-318 was tested prior to removing AFW pump P-319 from service for breaker maintenance.
2. When pump P-318 was shut down, the manual/overspeed trip was not properly reset. This resulted in both AFW pumps being unavailable for seven hours.

Corrective action:

The pump trip was correctly reset, the pump was tested, and returned to service.

Design purpose of failed system or component:

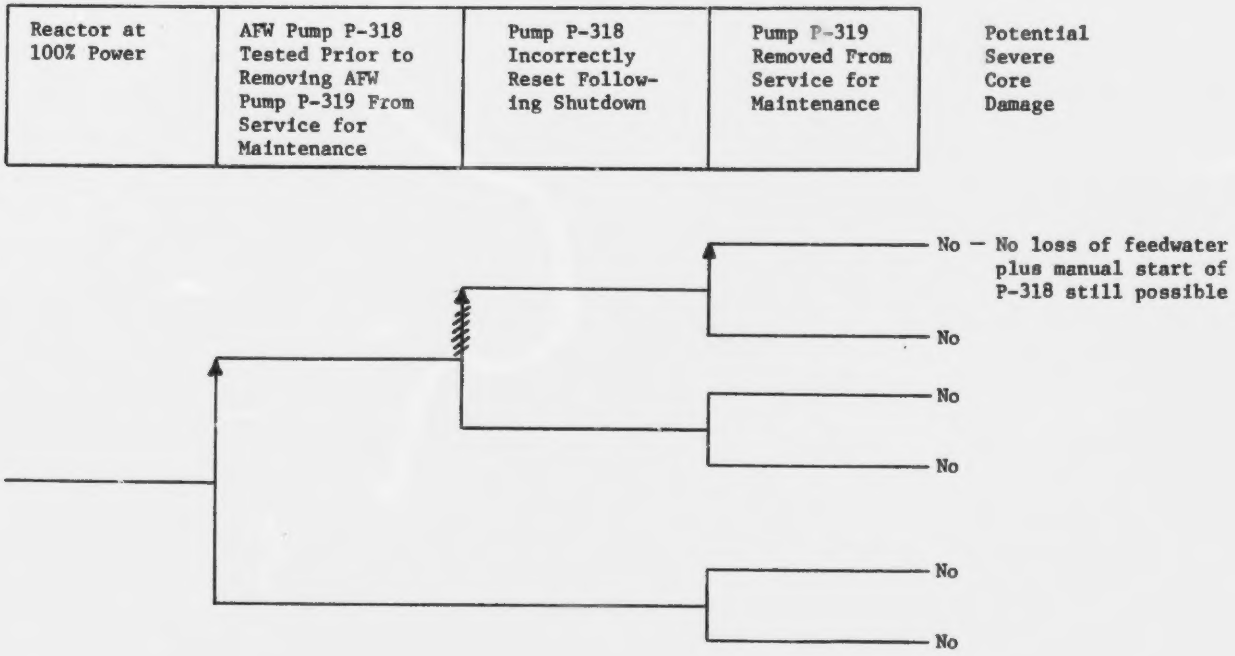
The auxiliary feedwater system provides steam generator cooling in the event the main feedwater system is unavailable.

Unavailability of system per WASH 1400:* AFW (start + 8 hours): 3.7×10^{-5}

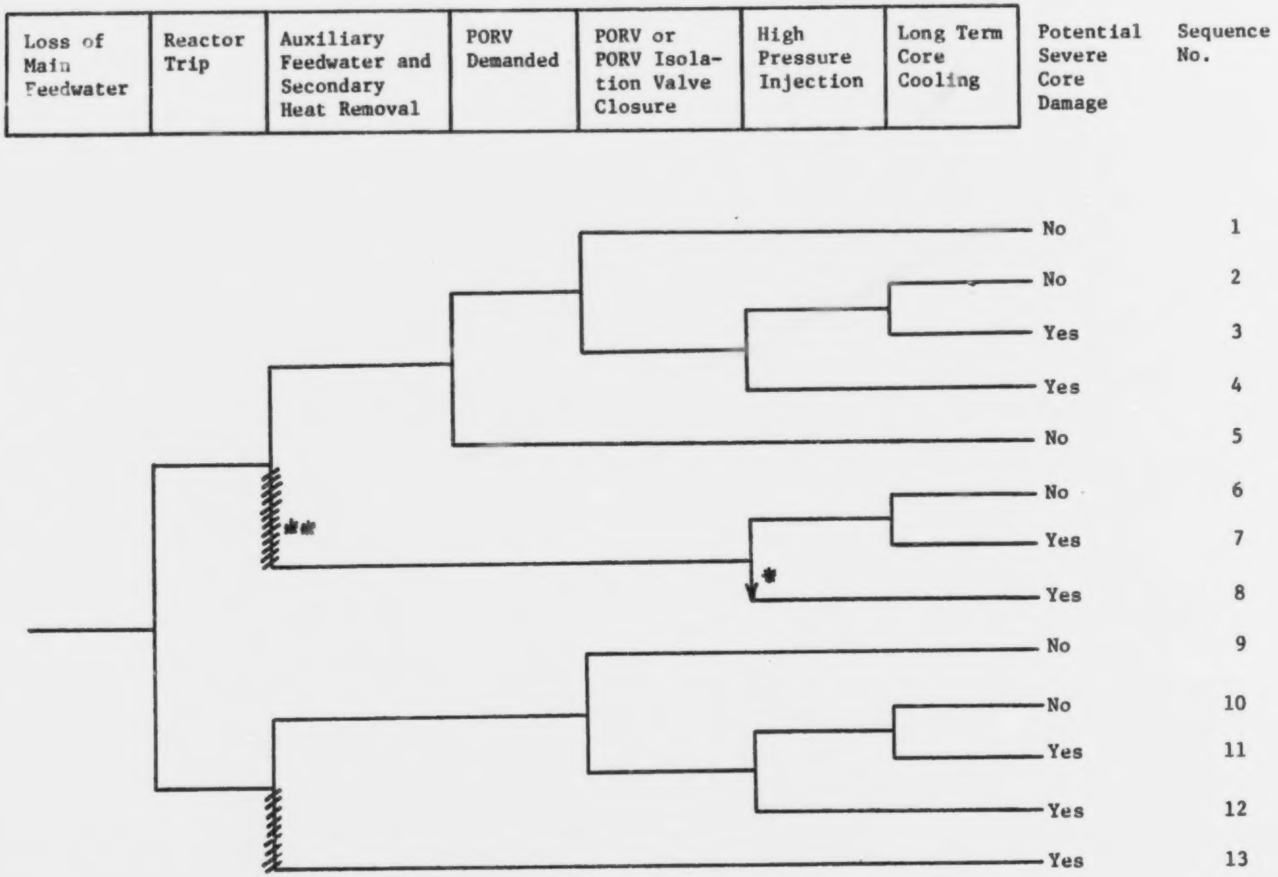
Unavailability of component per WASH 1400:* Pump, failure to start: 1×10^{-3}/D

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor at 100% Power
AFW Pump P-318 Tested Prior to Removing AFW Pump P-319 From Service for Maintenance
Pump P-318 Incorrectly Reset Following Shutdown
Pump P-319 Removed From Service for Maintenance
Potential Severe Core Damage
No — No loss of feedwater plus manual start of P-318 still possible
No
No
No
No
No

NSIC 123118 — Actual Occurrence for Both AFW Pumps Unavailable at Rancho Seco


Loss of Main Feedwater
Reactor Trip
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
Yes 3
Yes 4
No 5
No 6
Yes 7
Yes 8
No 9
No 10
Yes 11
Yes 12
Yes 13
**
*

NSIC 123118 — Sequence of Interest for Both AFW Pumps Unavailable at Rancho Seco

*Use of HPI following failure of AFW not included in operating procedures

**Operator action required for degraded

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 123118

DATE OF LER: March 9, 1977

DATE OF EVENT: February 18, 1977

SYSTEM INVOLVED: Auxiliary feedwater

COMPONENT INVOLVED: Auxiliary feedwater pump

CAUSE: Improperly reset manual/overspeed trip mechanism

SEQUENCE OF INTEREST: Loss of main feedwater

ACTUAL OCCURRENCE: Unavailability of both auxiliary feedwater pumps while at power

REACTOR NAME: Rancho Seco

DOCKET NUMBER: 312

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 918 MWe

REACTOR AGE: 2.4 yr

VENDOR: B&W

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Sacramento Municipal Utility District

LOCATION: 25 miles SE of Sacramento, California

DURATION: 7 hours

PLANT OPERATING CONDITION: 100% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________________

DISCOVERY METHOD: Operator observation

COMMENT:

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 123150

Date: March 10, 1977

Title: Excess Cooldown Rate Results from Loss of ICS Power at Crystal River 3

The failure sequence was:

1. Inverter B output diode failed in the B a.c. inverter, resulting in loss of vital bus B.
2. This resulted in a loss of power to the ICS and consequent reactor trip, turbine trip, and 50% opening of the atmospheric dump valves.
3. Main feedwater was lost due to loss of condenser vacuum.
4. Emergency feedwater was used for reactor cooling.

Corrective action;

1. The B inverter output diode was replaced and the inverter restored to operation.

Design purpose of failed system or component:

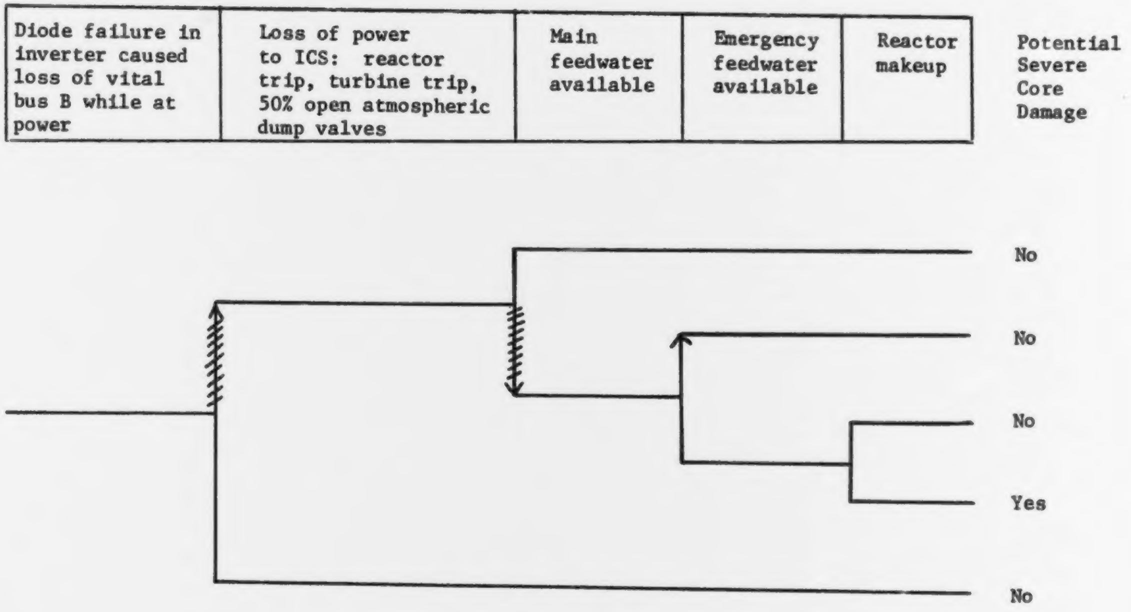
1. The inverter provides battery-backed A.C. power for instrumentation loads.

Unavailability of system per WASH 1400:* vital bus: 3×10^{-6}/hr.

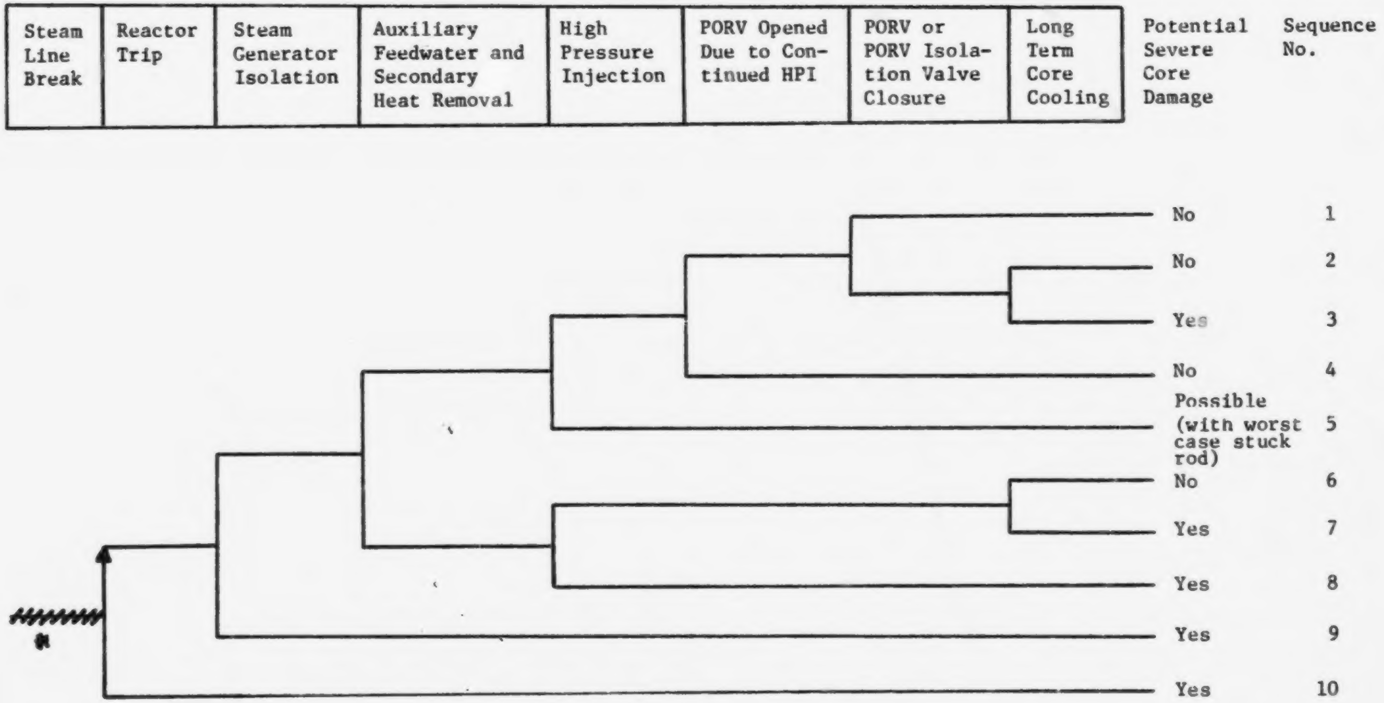
Unavailability of component per WASH 1400:* power supplies: 3×10^{-6}/hr.

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Diode failure in inverter caused loss of vital bus B while at power
Loss of power to ICS: reactor trip, turbine trip, 50% open atmospheric dump valves
Main feedwater available
Emergency feedwater available
Reactor makeup
Potential Severe Core Damage
No
No
No
Yes
No

NSIC 123150 — Actual Occurrence for Excess Cooldown Rate Results from Loss of ICS Power at Crystal River 3.


Steam Line Break
Reactor Trip
Steam Generator Isolation
Auxiliary Feedwater and Secondary Heat Removal
High Pressure Injection
PORV Opened Due to Continued HPI
PORV or PORV Isolation Valve Closure
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
Yes 3
No 4
Possible (with worst case stuck rod) 5
No 6
Yes 7
Yes 8
Yes 9
Yes 10
*

NSIC 123150 — Sequence of Interest for Excess Cooldown Rate Results from Loss of ICS Power at Crystal River 3

*small

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 123150

DATE OF LER: March 10, 1977

DATE OF EVENT: March 3, 1977

SYSTEM INVOLVED: Vital power supply, Integrated Control System

COMPONENT INVOLVED: Inverter

CAUSE: Failed output diode

SEQUENCE OF INTEREST: Stuck open dump valves (Small steam line break)

ACTUAL OCCURRENCE: 50% open atmospheric dump valves

REACTOR NAME: Crystal River 3

DOCKET NUMBER: 50-302

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 825 MWe

REACTOR AGE: 0.25 yr

VENDOR: Babcock & Wilcox

ARCHITECT-ENGINEERS: Gilbert Assoc.

OPERATORS: Florida Power Corp.

LOCATION: 7 miles NW of Crystal River, Fla.

DURATION: N/A

PLANT OPERATING CONDITION: At 40% power.

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; (c) made inoperable; [✓ (d) failed in an unsafe manner]

DISCOVERY METHOD: Event while operating

COMMENT: —

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 124201

Date: March 22, 1977

Title: Diesel Generator Field Fails to Flash at Dresden 3

The failure sequence was:

1. Diesel generator #3 was undergoing testing after maintenance when its field failed to flash.
2. The field flashed on the second attempt, but failed on the third try. The problem was traced to an intermittent capacitor short (this problem doesn't appear to be related to the maintenance activity).
3. A subsequent test of the swing diesel (2/3) also proved unsuccessful. It was successfully started 45 minutes later. The problem was tranced to air starter motors.

Corrective action:

1. The failed capacitor in unit 3 was replaced.
2. The air starter motors in unit 2/3 were replaced.
3. Procedural changes were made which require the capacitors and the air starter motors to be inspected and tested after maintenance.

Design purpose of failed system or component:

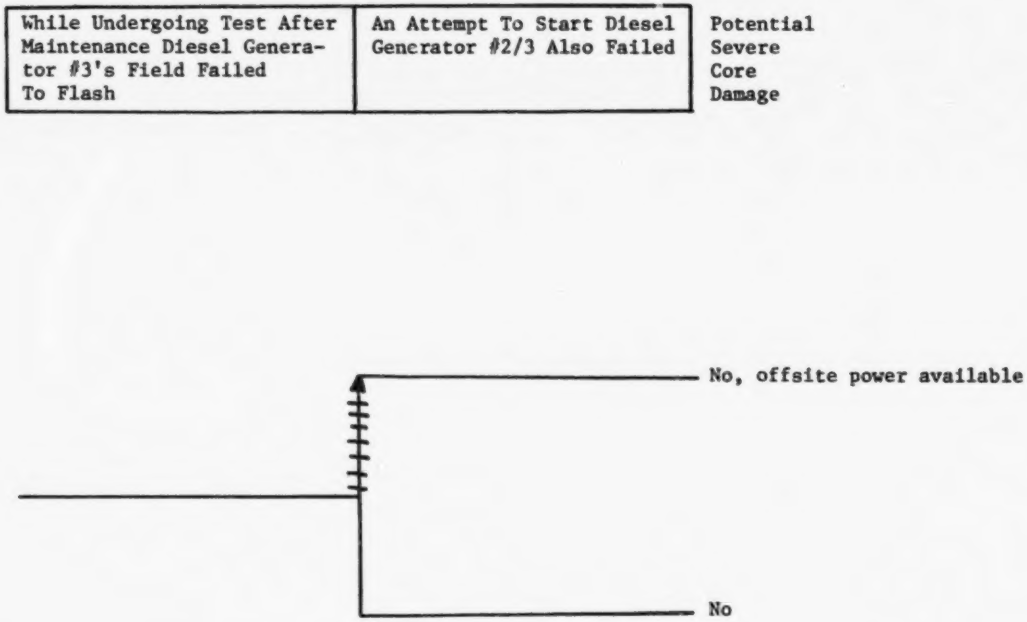
1. Emergency power provides a.c. power when offsite power is unavailable.
2. The diesel generator produce the emergency a.c. power.

Unavailability of system per WASH 1400:* Emergency Electric Power: 10^{-3}/D

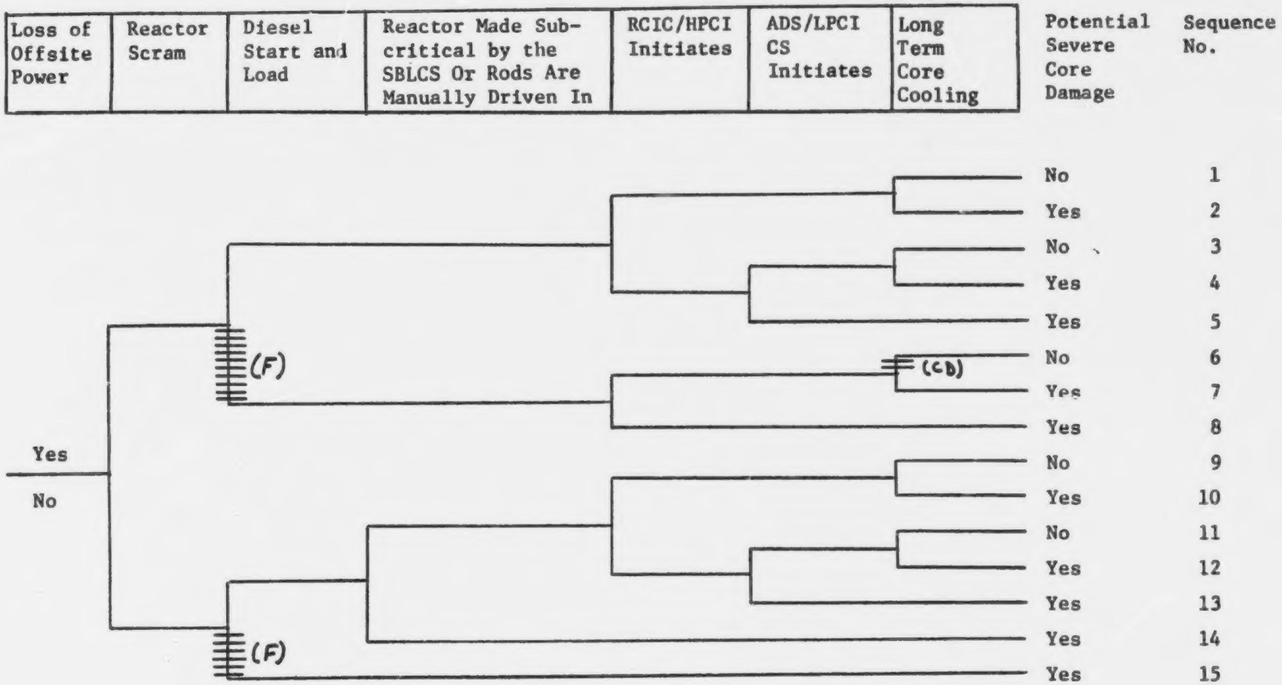
Unavailability of component per WASH 1400:* Diesel Generator failure to start: 3×10^{-2}/D

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


While Undergoing Test After Maintenance Diesel Generator #3's Field Failed To Flash
An Attempt To Start Diesel Generator #2/3 Also Failed
Potential Severe Core Damage
No, offsite power available
No

NSIC 124201 — Actual Occurrence of a Diesel Generator's Field Fails to Flash at Dresden 3


Loss of Offsite Power
Reactor Scram
Diesel Start and Load
Reactor Made Subcritical by the SBLCS Or Rods Are Manually Driven In
RCIC/HPCI Initiates
ADS/LPCI CS Initiates
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
(F)
(CD)
(F)
No 1
Yes 2
No 3
Yes 4
Yes 5
No 6
Yes 7
Yes 8
No 9
Yes 10
No 11
Yes 12
Yes 13
Yes 14
Yes 15

NSIC 124201 — Sequence of Interest for a Diesel Generator's Field Fails to Flash at Dresden 3

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 124201

DATE OF LER: April 20, 1977

DATE OF EVENT: March 22, 1977

SYSTEM INVOLVED: Emergency Electric Power

COMPONENT INVOLVED: Diesel Genertor

CAUSE: Electrical short of a capacitor, mechanical failure of diesel generator

SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE: Diesel generator field fails to flash at Dresden 3

REACTOR NAME: Dresden 3

DOCKET NUMBER: 50-249

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 809 MWe

REACTOR AGE: 7.2 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: Sargent & Lundy

OPERATORS: Commonwealth Edison

LOCATION: Nine miles E of Morris, Ill.

DURATION: 12 (a) hours

PLANT OPERATING CONDITION: 57% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________________

DISCOVERY METHOD: testing

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 124222

Date: March 28, 1977

Title: Six Main Steam Relief Valves Fail to Lift Properly at Duane Arnold

The failure sequence was:

1. During bench testing of six main steam relief valves failed to lift at the required pressure, four failed to open and the remaining two lifted at elevated pressures.

Corrective action:

The problem was being investigated, however at the time of this report the cause hadn't been resolved.

Design purpose of failed system or component:

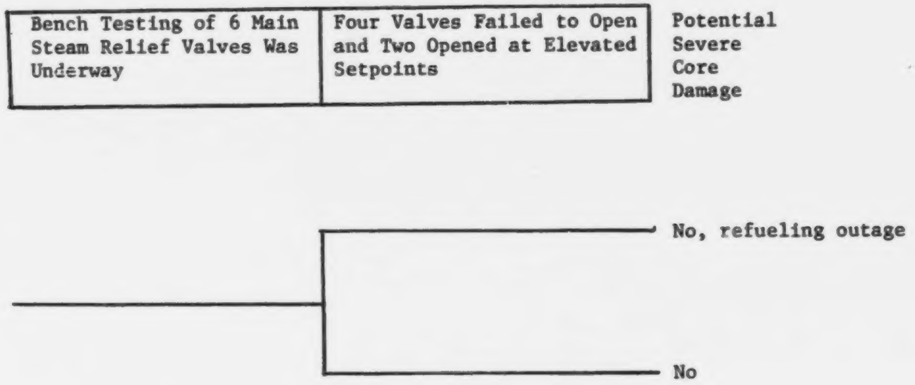
The steam relief valves allow for pressure control.

Unavailability of system per WASH 1400:[*] -

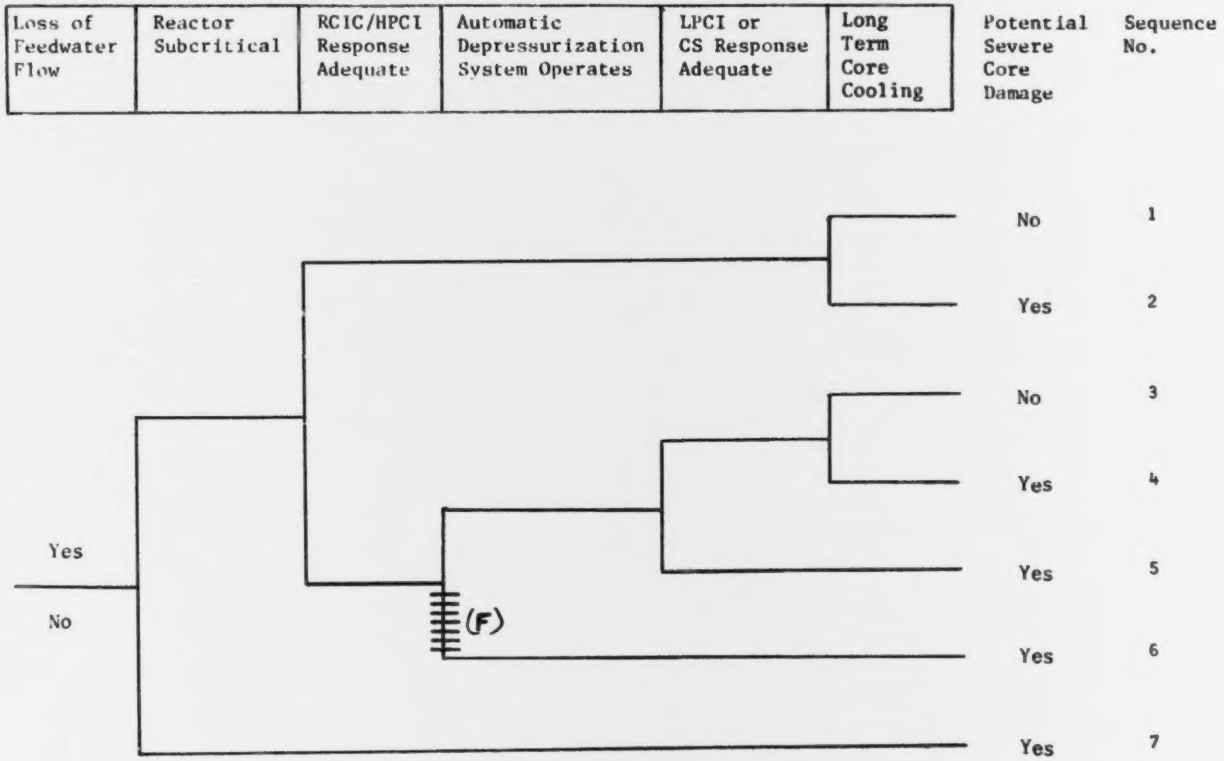
Unavailability of component per WASH 1400:[*] 1.0×10^{-5}/D

---

[*] Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Bench Testing of 6 Main Steam Relief Valves Was Underway
Four Valves Failed to Open and Two Opened at Elevated Setpoints
Potential Severe Core Damage
No, refueling outage
No

NSIC 124222 — Actual Occurrence of Six Main Steam Relief Valves Fail to Lift Properly at Duane Arnold


Loss of Feedwater Flow
Reactor Subcritical
RCIC/HPCI Response Adequate
Automatic Depressurization System Operates
LPCI or CS Response Adequate
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
(F)
No 1
Yes 2
No 3
Yes 4
Yes 5
Yes 6
Yes 7

NSIC 124222 — Sequence of Interest for Six Main Steam Relief Valves Fail to Lift Properly at Duane Arnold

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 124222

DATE OF LER: April 2, 1977

DATE OF EVENT: March 28, 1977

SYSTEM INVOLVED: ADS

COMPONENT INVOLVED: Relief Valves

CAUSE: Unknown

SEQUENCE OF INTEREST: LOFW

ACTUAL OCCURRENCE: Six Main Steam Relief Valves Fail to Lift Properly at Duane Arnold.

REACTOR NAME: Duane Arnold.

DOCKET NUMBER: 331

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 821 MWe

REACTOR AGE: 3.0 yr

VENDOR: G.E.

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Iowa Electric Power and Light

LOCATION: Cedar Rapids, Iowa

DURATION: 4883 (a) hours

PLANT OPERATING CONDITION: 0%

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ______________________

DISCOVERY METHOD: Surveillance Testing

COMMENT: Four valves failed to open during bench test.

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 124920

Date: April 16, 1977

Title: Excessive Cooldown Rate Experienced During Testing at Crystal River 3

The failure sequence was:

1. During shutdown testing outside the control room, an out-of-calibration transmitter prevented complete closure of the startup feedwater control valves for steam generators A and B, resulting in a steam generator overfill condition.
2. Feed pump throttling was unsuccessfully attempted.
3. The steam generator atmospheric dump valves and steam generator drain valves were opened to mitigate the overfill condition.
4. RCS makeup was provided from the BWST.

Corrective action;

1. The out-of-calibration pressure transmitter was recalibrated.
2. Test and operating procedures were revised to require the feed pumps to be tripped to prevent overfilling the steam generators.

Design purpose of failed system or component:

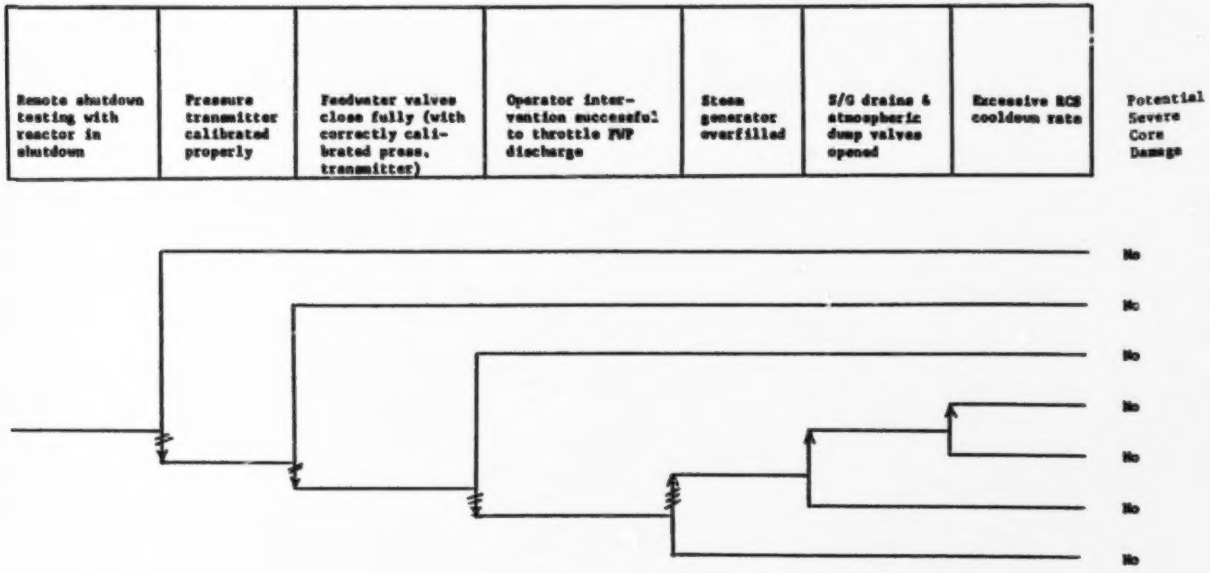
1. The function of the pressure transmitter cannot be determined from available information.

Unavailability of system per WASH 1400:* —

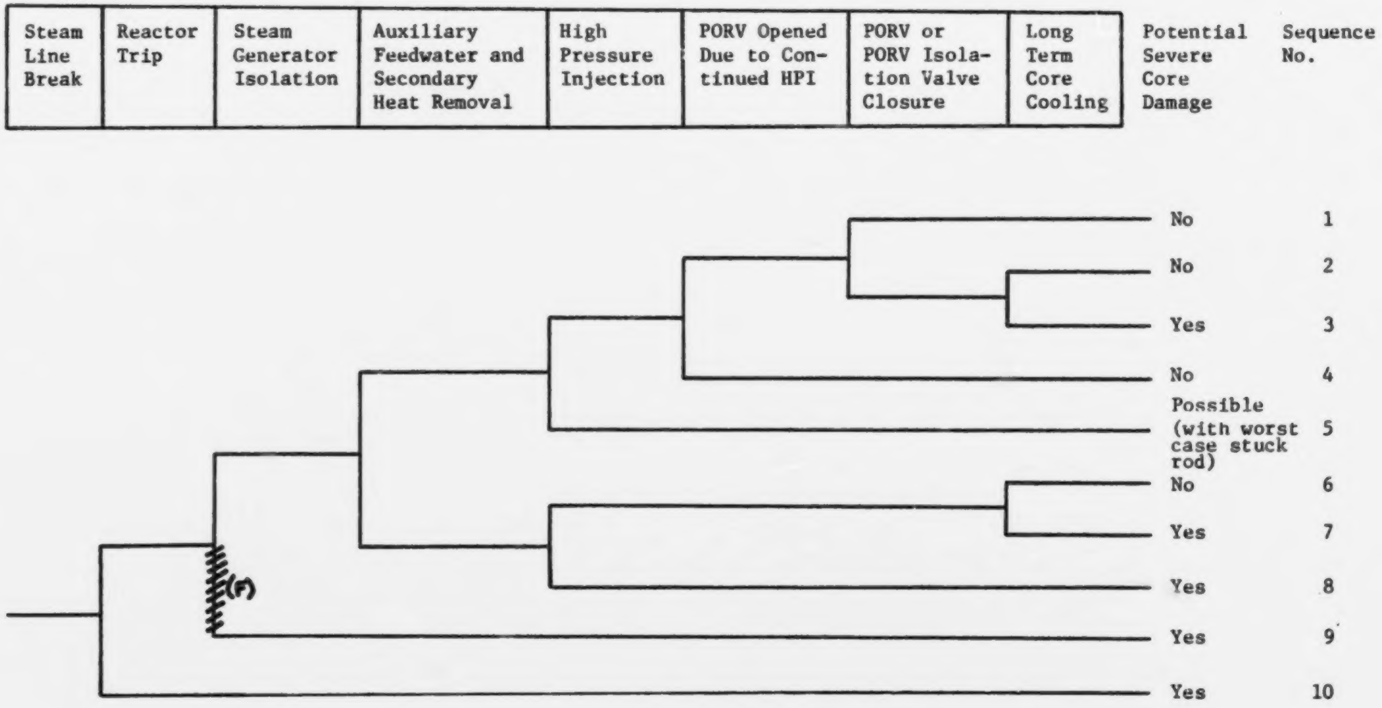
Unavailability of component per WASH 1400:* Instrumentation, calibration shift 3×10^{-5}/hr.

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Remote shutdown testing with reactor in shutdown
Pressure transmitter calibrated properly
Feedwater valves close fully (with correctly calibrated press. transmitter)
Operator intervention successful to throttle FWP discharge
Steam generator overfilled
S/G drains & atmospheric dump valves opened
Excessive RCS cooldown rate
Potential Severe Core Damage
No
No
No
No
No
No
No

NSIC 124920 — Actual Occurrence for Excessive Cooldown Rate Experienced During Testing at Crystal River 3


Steam Line Break
Reactor Trip
Steam Generator Isolation
Auxiliary Feedwater and Secondary Heat Removal
High Pressure Injection
PORV Opened Due to Continued HPI
PORV or PORV Isolation Valve Closure
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
(F)
No 1
No 2
Yes 3
No 4
Possible (with worst case stuck rod) 5
No 6
Yes 7
Yes 8
Yes 9
Yes 10

NSIC 124920 — Sequence of Interest for Excessive Cooldown Rate Experienced During Testing at Crystal River 3

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 124920

DATE OF LER: April 21, 1977

DATE OF EVENT: April 16, 1977

SYSTEM INVOLVED: Feedwater

COMPONENT INVOLVED: Pressure transmitter

CAUSE: Out-of-calibration

SEQUENCE OF INTEREST: Steam line break

ACTUAL OCCURRENCE: Steam generator overfill

REACTOR NAME: Crystal River 3

DOCKET NUMBER: 50-302

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 825 MWe

REACTOR AGE: 0.3 yr

VENDOR: Babcock & Wilcox

ARCHITECT-ENGINEERS: Gilbert Assoc.

OPERATORS: Florida Power Corp.

LOCATION: 7 miles NW of Crystal River, Fla.

DURATION: 360(a) hours

PLANT OPERATING CONDITION: Hot shutdown

SAFETY FEATURE TYPE OF FAILURE: [✓ (a)] inadequate performance; (b) failed to start;
(c) made inoperable; (d) ________________

DISCOVERY METHOD: During testing.

COMMENT: –

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 125106

Date: May 20, 1977

Title: Both SI Pumps and One Charging Pump Out of Service During Heatup at Salem 1

The failure sequence was:

1. During plant heatup, both SI pumps plus one centrifugal charging pump were not cleared from cold shutdown overpressure control.

Corrective action:

1. The overpressure control restrictions were cleared upon discovery.

Design purpose of failed system or component:

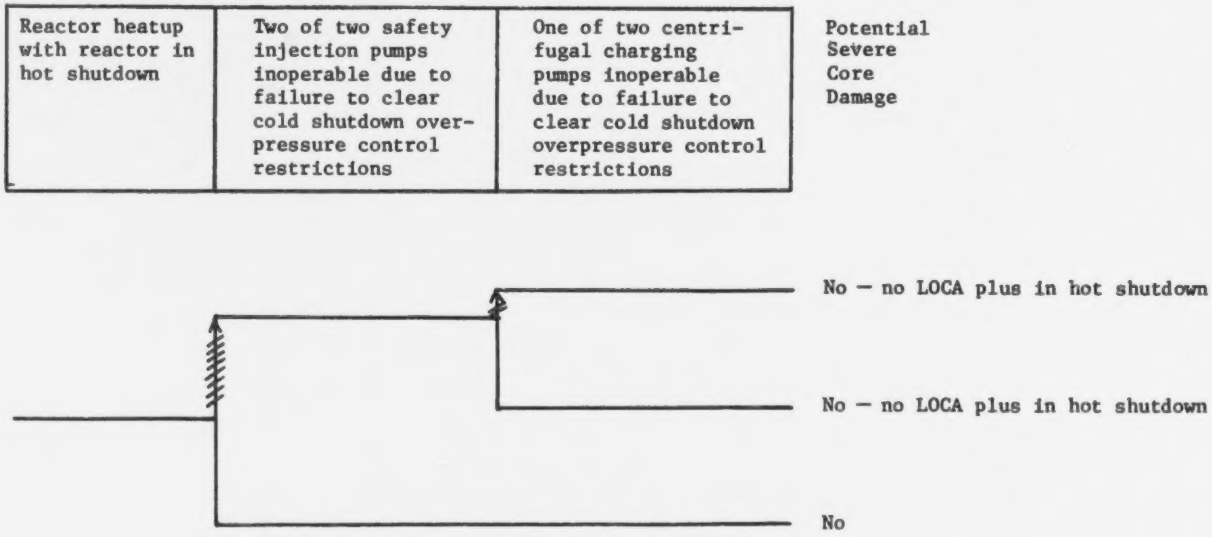
1. The Safety Injection pumps and the centrifugal charging pumps provide high head safety injection during a small LOCA.

Unavailability of system per WASH 1400:* High Pressure Injection: 8.6×10^{-3}/D

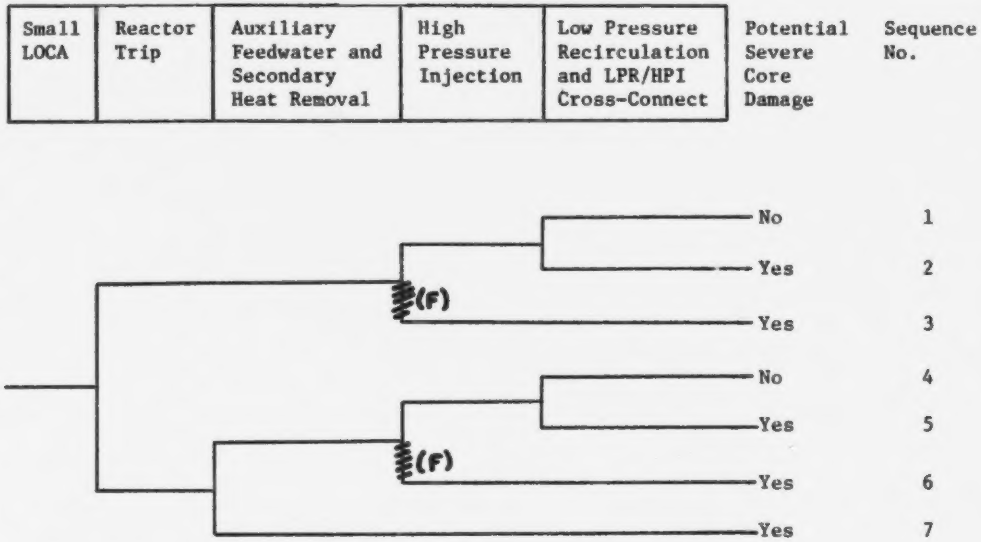
Unavailability of component per WASH 1400:* operator error, failure to follow a step within a procedure: 3×10^{-3}/D

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor heatup with reactor in hot shutdown
Two of two safety injection pumps inoperable due to failure to clear cold shutdown overpressure control restrictions
One of two centrifugal charging pumps inoperable due to failure to clear cold shutdown overpressure control restrictions
Potential Severe Core Damage
No — no LOCA plus in hot shutdown
No — no LOCA plus in hot shutdown
No

NSIC 125106 — Actual Occurrence for Both SI Pumps and One Charging Pump Out of Service During Heatup at Salem 1


Small LOCA
Reactor Trip
Auxiliary Feedwater and Secondary Heat Removal
High Pressure Injection
Low Pressure Recirculation and LPR/HPI Cross-Connect
Potential Severe Core Damage
Sequence No.
(F)
(F)
No 1
Yes 2
Yes 3
No 4
Yes 5
Yes 6
Yes 7

NSIC 125106 — Sequence of Interest for Both SI Pumps and One Charging Pump Out of Service During Heatup at Salem 1

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 125106

DATE OF LER: May 20, 1977

DATE OF EVENT: May 6, 1977

SYSTEM INVOLVED: Safety Injection, Charging

COMPONENT INVOLVED: SI pumps, charging pump

CAUSE: Failure to clear cold-shutdown overpressure control restrictions, human error.

SEQUENCE OF INTEREST: Small LOCA

ACTUAL OCCURRENCE: failure to clear overpressure restrictions during heatup

REACTOR NAME: Salem 1

DOCKET NUMBER: 50-272

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 1090 MWe

REACTOR AGE: 0.8 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Utility

OPERATORS: Public Service Electric & Gas Co.

LOCATION: 20 miles south of Wilmington, DE

DURATION: 3.4 hours

PLANT OPERATING CONDITION: hot shutdown

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ________________

DISCOVERY METHOD: plant equipment status review

COMMENT: –

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 125563

Date: May 16, 1977

Title: Offsite Power Lost at St. Lucie 1

The failure sequence was:

1. Fifteen minutes after a manual reactor trip due to grid disturbances, a system undervoltage condition occurred which caused the reactor coolant pumps to trip.
2. Both diesel-generators started and loaded.
3. The reactor was cooled down using natural circulation until off-site power was restored (approximately 20 minutes).
4. One hour later, off-site power was again lost and natural circulation cooldown was used until off-site power was restored (one to one-and-a-half hours later).

Corrective action;

None

Design purpose of failed system or component:

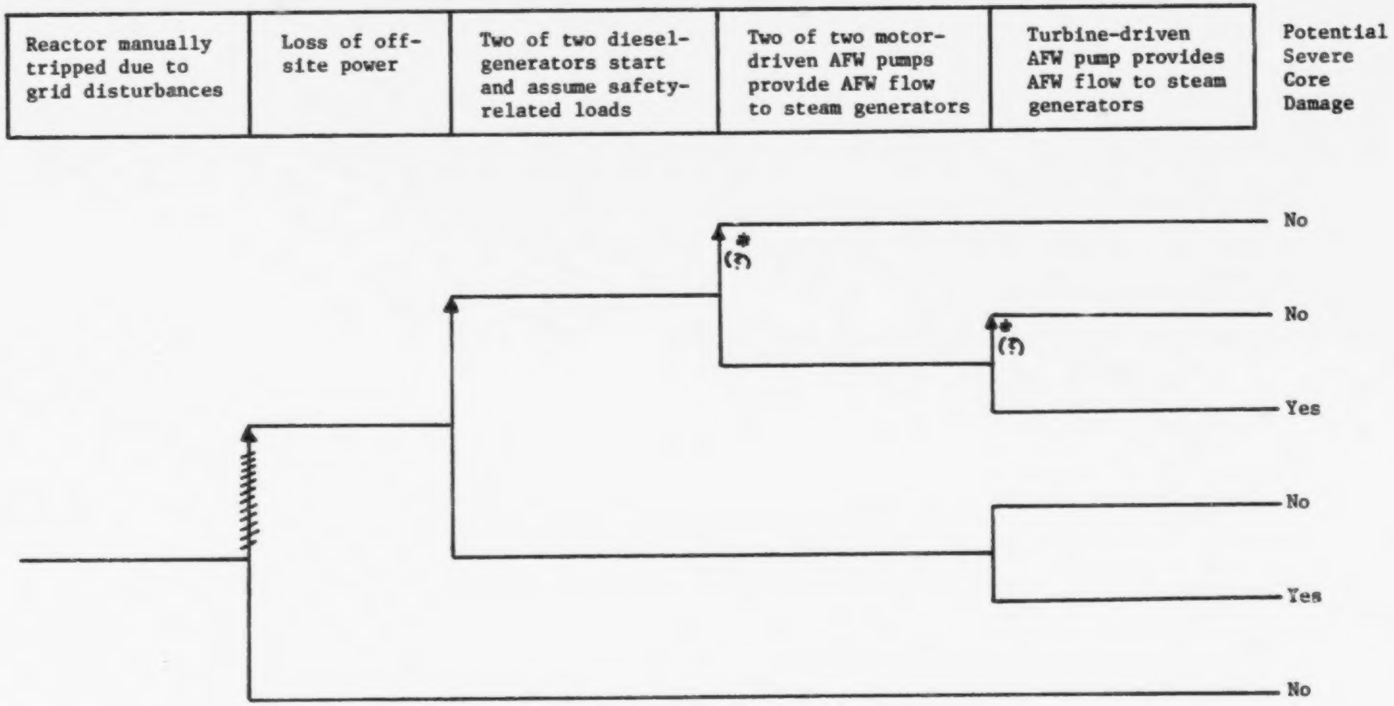
1. Off-site power provides the preferred source of electric power to safety-related equipment when the unit generator is not in operation.

Unavailability of system per WASH 1400:* loss of offsite power: 2×10^{-5}/hr

Unavailability of component per WASH 1400:* –

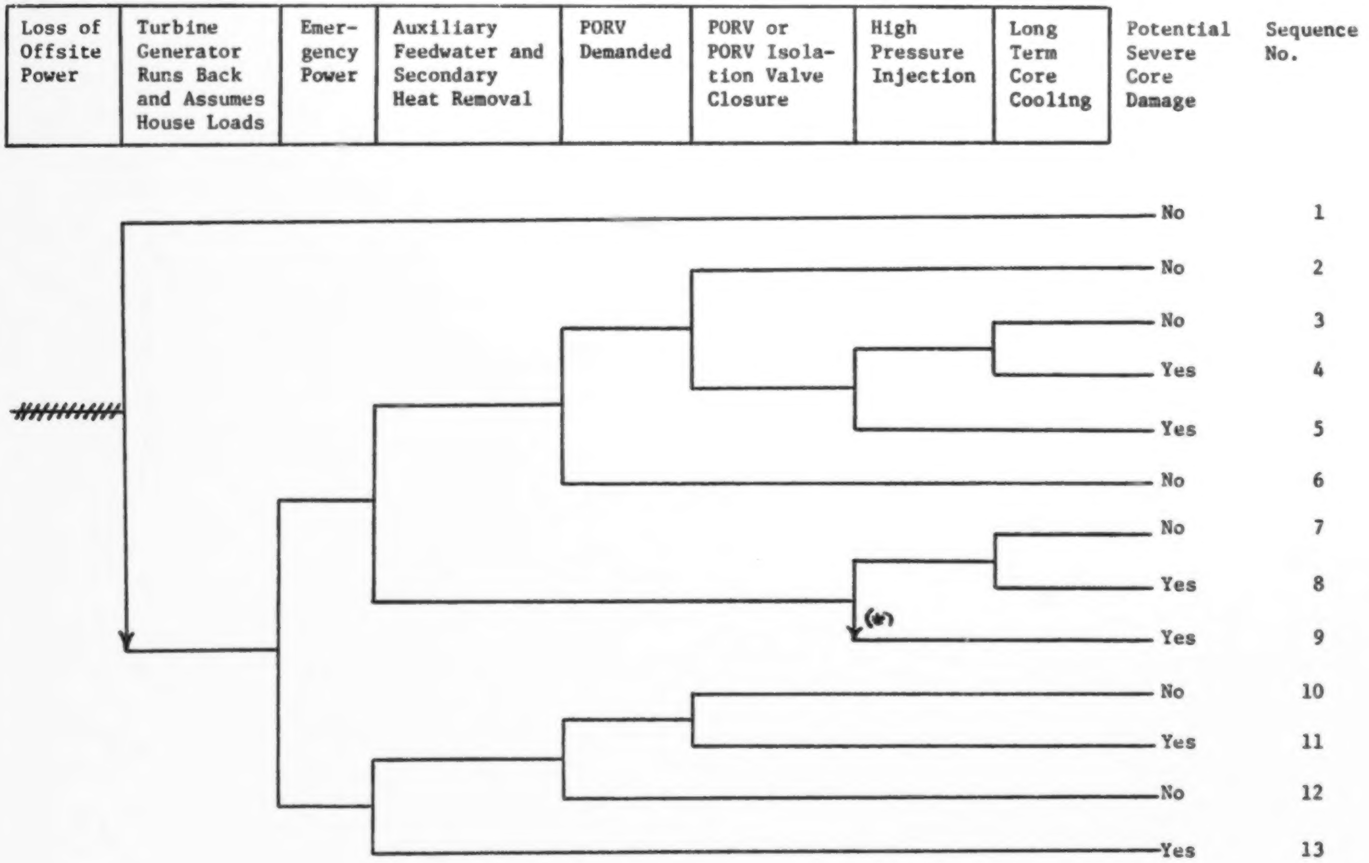
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor manually tripped due to grid disturbances
Loss of off-site power
Two of two diesel-generators start and assume safety-related loads
Two of two motor-driven AFW pumps provide AFW flow to steam generators
Turbine-driven AFW pump provides AFW flow to steam generators
Potential Severe Core Damage
*
(?)
*
(?)
No
No
Yes
No
Yes
No

NSIC 125563 — Actual Occurrence for Off-Site Power Lost at St. Lucie 1

*specific pumps providing AFW flow not specified.


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
(*)
Yes 9
No 10
Yes 11
No 12
Yes 13

NSIC 125563 — Sequence of Interest for Off-Site Power Lost at St. Lucie 1

*not included in mitigation procedures.

365

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 125563

DATE OF LER: June 16, 1977

DATE OF EVENT: May 16, 1977

SYSTEM INVOLVED: off-site power

COMPONENT INVOLVED: unknown

CAUSE: unknown

SEQUENCE OF INTEREST: loss of offsite power

ACTUAL OCCURRENCE: reactor trip with loss of offsite power

REACTOR NAME: St. Lucie 1

DOCKET NUMBER: 50-335

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 802 MWe

REACTOR AGE: 1.1 yr

VENDOR: Combustion Engineering

ARCHITECT-ENGINEERS: Ebasco

OPERATORS: Florida Power & Light Co.

LOCATION: 12 miles SE of Fort Pierce, Fla.

DURATION: N/A

PLANT OPERATING CONDITION: tripped prior to loss of off-site power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ________________

DISCOVERY METHOD: during operation

COMMENT: LER does not specify which AFW system components were used to provide core cooling.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 127010

Date: June 30, 1977

Title: Failure of two Diesel Generators at Dresden 3

The failure sequence was:

1. Diesel 3 was out of service for maintenance.
2. Diesel 2/3 failed to start during an operability test.
3. The injector control lever was exercised and the diesel successfully started.

Corrective action:

3. The injector control lever was manually exercised.

Design purpose of failed system or component:

Emergenc Power provides a.c. power when offsite power is unavailable.

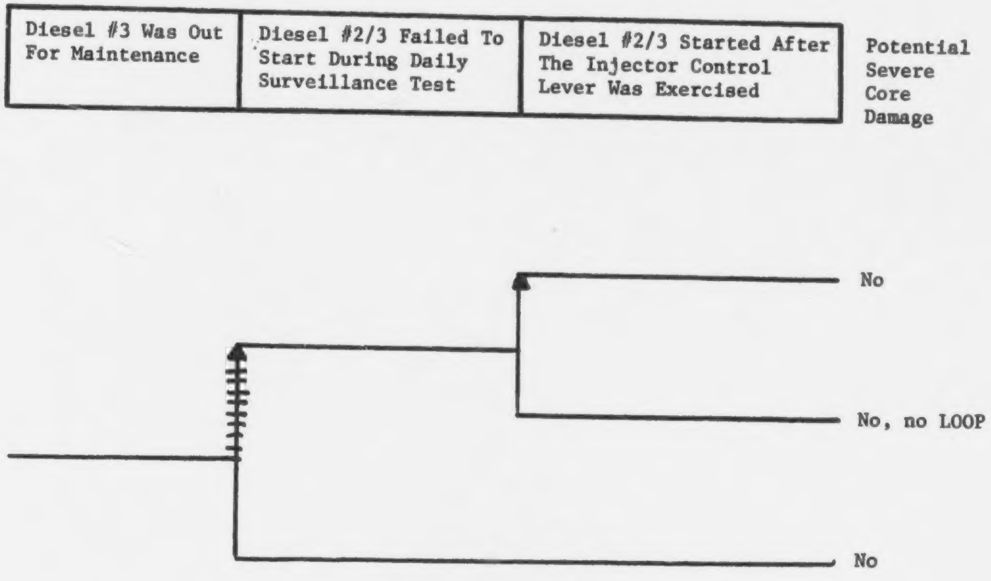
Unavailability of system per WASH 1400:* Emergency Power: 1×10^{-3}/D

Unavailability of component per WASH 1400:* Diesel Generator Failure to start: 3×10^{-2}/D

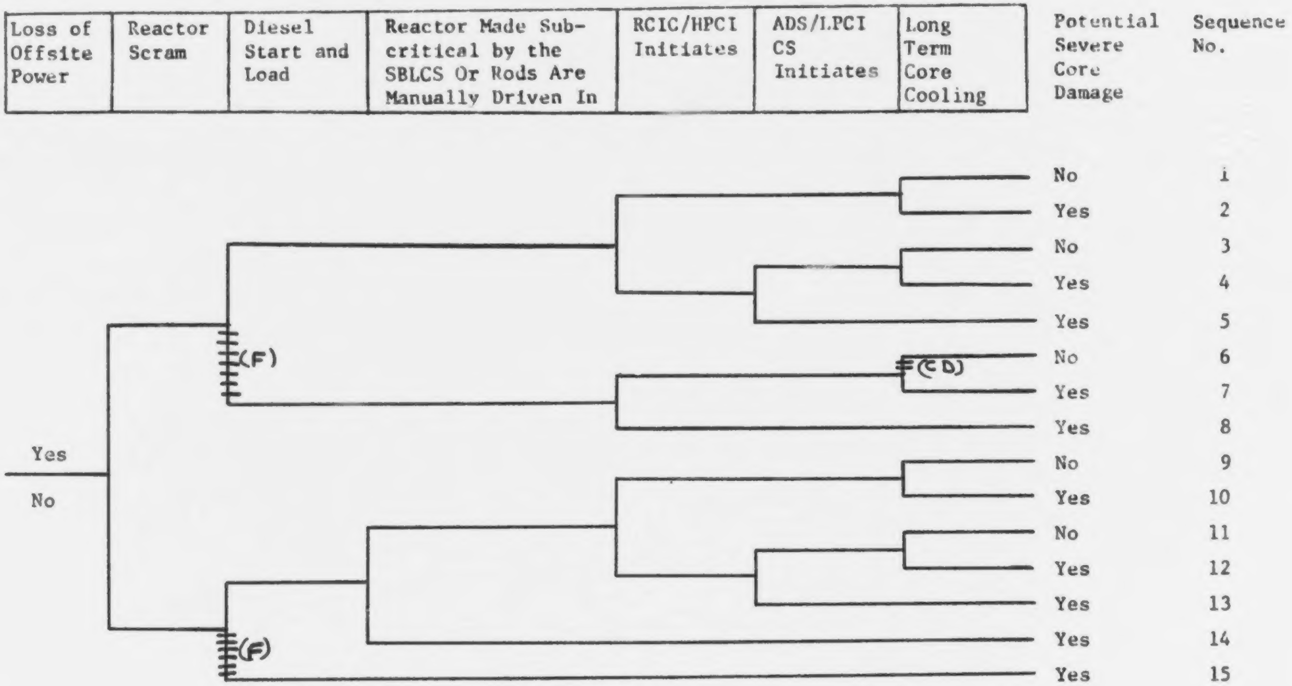
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

BEST DOCUMENT AVAILABLE


Diesel #3 Was Out For Maintenance
Diesel #2/3 Failed To Start During Daily Surveillance Test
Diesel #2/3 Started After The Injector Control Lever Was Exercised
Potential Severe Core Damage
No
No, no LOOP
No

NSIC 127010 — Actual Occurrence of Diesel Generator Fails to Start at Dresden 3


Loss of Offsite Power
Reactor Scram
Diesel Start and Load
Reactor Made Sub-critical by the SBLCS Or Rods Are Manually Driven In
RCIC/HPCI Initiates
ADS/LPCI CS Initiates
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
(F)
(CD)
(F)
No 1
Yes 2
No 3
Yes 4
Yes 5
No 6
Yes 7
Yes 8
No 9
Yes 10
No 11
Yes 12
Yes 13
Yes 14
Yes 15

NSIC 127010 — Sequence of Interest for Failure of Two Diesel Generators at Dresden 2

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 127010

DATE OF LER: July 14, 1977

DATE OF EVENT: June 30, 1977

SYSTEM INVOLVED: Emergency Power

COMPONENT INVOLVED: Diesel Generators

CAUSE: Mechanical failure and maintenance

SEQUENCE OF INTEREST: LOOP

ACTUAL OCCURRENCE: Failure of two Diesel Generators

REACTOR NAME: Dresden 3

DOCKET NUMBER: 50-249

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 794 MWe

REACTOR AGE: 6.4 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: Sargent and Lundy

OPERATORS: Commonwealth Edison

LOCATION: 9 miles East of Morris, Ill.

DURATION: 12 (a) hours

PLANT OPERATING CONDITION: 74% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performace; (b) failed to start; [✓ (c)] made inoperable; (d)______________________

DISCOVERY METHOD: Test

COMMENT:

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: **127384**

Date: **July 25, 1977**

Title: **Dummy Loads Installed in All Safety Systems at Zion 2**

The failure sequence was:

1. With the reactor in hot shutdown, dummy loads were installed on all pressurizer level, pressurizer pressure, reactor coolant loop flow, and steam generator level instrumentation during a periodic reactor protection system logic test. This left the operator with no real indication of these parameters and provided incorrect inputs to the control functions associated with these parameters. These dummy inputs also negated all related inputs for safety injection actuation. (Actuation on high containment pressure and steam generator differential pressure still existed.)
2. The dummy signal for pressurizer level was set 2% higher than the actual pressurizer level at the time of the test, which resulted in a control signal which reduced charging pump flow and subsequently began reducing pressurizer level. (Letdown flow remained constant.)
3. Soon afterwards, the unit operator noticed a reactor coolant pump seal Δp alarm (caused by the decreased charging pump flow) and attempted to restore proper Δp by (cont)

Corrective action:

1. The pressurizer heaters were checked for proper current and resistance; all indications were normal.
2. The test procedure has been changed to eliminate the need for dummy signals. Other protection and safeguards periodic tests were reviewed to minimize and further control the use of dummy signals.

Design purpose of failed system or component:

1. The affected instrumentation provides trip inputs for reactor protection and safety injection actuation and control inputs for reactor auxiliary system operation.

Unavailability of system per WASH 1400:* RPS: 3.6×10^{-5}/D

Safety Injection Control System: 9.9×10^{-5}/D

Unavailability of component per WASH 1400:* General operator error: 3×10^{-3}

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

PRECURSOR DESCRIPTION AND DATA NSIC Accession Number: 127384

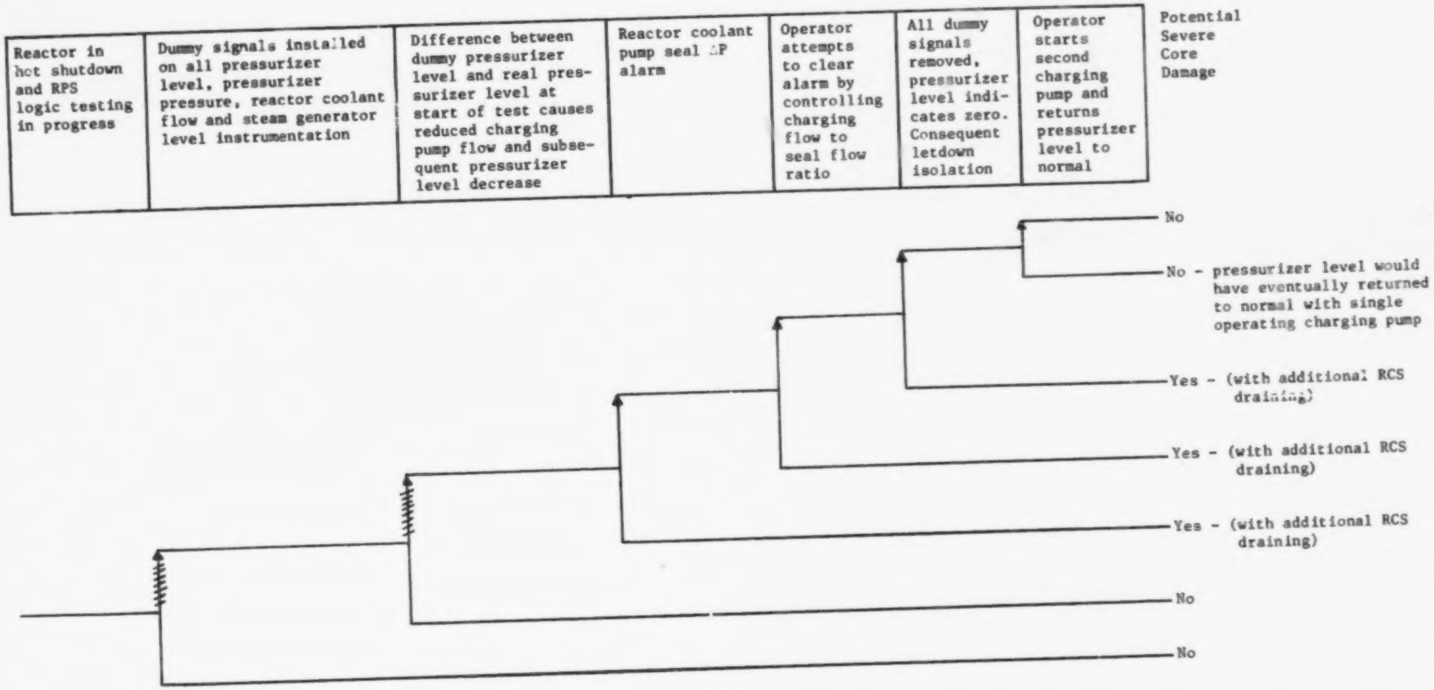
The failure sequence was: (cont)

adjusting the charging flow to seal flow ratio.

4. Thirty-two minutes after the start of the Logic test the operator requested the dummy signals be removed from the steam generator level and pressurizer pressure instrumentation. Forty minutes after the test began all remaining dummy loads were removed.

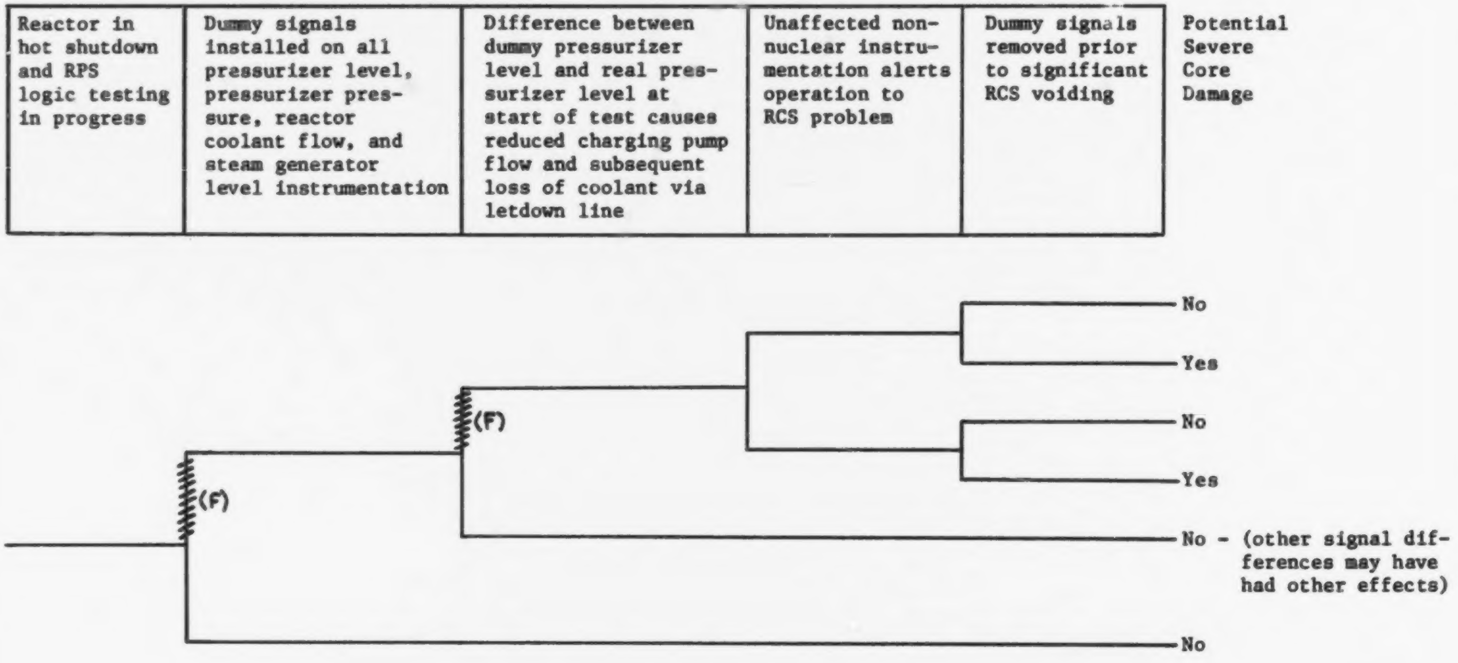
5. When actual indication was restored, the pressurizer level indicated zero. This isolated letdown and tripped the pressurizer heaters. (Subsequent calculations indicated the pressurizer level had dropped to within the surge line.)

6. The operator started a second charging pump and, within 30 minutes, the pressurizer level returned to normal.


Reactor in hot shutdown and RPS logic testing in progress
Dummy signals installed on all pressurizer level, pressurizer pressure, reactor coolant flow and steam generator level instrumentation
Difference between dummy pressurizer level and real pressurizer level at start of test causes reduced charging pump flow and subsequent pressurizer level decrease
Reactor coolant pump seal ΔP alarm
Operator attempts to clear alarm by controlling charging flow to seal flow ratio
All dummy signals removed, pressurizer level indicates zero. Consequent letdown isolation
Operator starts second charging pump and returns pressurizer level to normal
Potential Severe Core Damage
No
No - pressurizer level would have eventually returned to normal with single operating charging pump
Yes - (with additional RCS draining)
Yes - (with additional RCS draining)
Yes - (with additional RCS draining)
No
No

NSIC 127384 — Actual Occurrence for Dummy Loads Installed on All Safety Systems at Zion 2

BEST DOCUMENT AVAILABLE


Reactor in hot shutdown and RPS logic testing in progress
Dummy signals installed on all pressurizer level, pressurizer pressure, reactor coolant flow, and steam generator level instrumentation
Difference between dummy pressurizer level and real pressurizer level at start of test causes reduced charging pump flow and subsequent loss of coolant via letdown line
Unaffected non-nuclear instrumentation alerts operation to RCS problem
Dummy signals removed prior to significant RCS voiding
Potential Severe Core Damage
(F)
(F)
No
Yes
No
Yes
No - (other signal differences may have had other effects)
No

NSIC 127384 — Sequence of Interest for Dummy Loads Installed on All Safety Systems at Zion 2

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 127384

DATE OF LER: July 25, 1977

DATE OF EVENT: July 12, 1977

SYSTEM INVOLVED: Reactor protection system, non-nuclear instrumentation, safety injection actuation.

COMPONENT INVOLVED: Pressurizer pressure and level, steam generator level, reactor coolant loop flow instrumentation.

CAUSE: Dummy signals installed on all this instrumentation, human error

SEQUENCE OF INTEREST: LOCA

ACTUAL OCCURRENCE: Dummy signals installed on instrumentation and subsequent loss of pressurizer level.

REACTOR NAME: Zion 2

DOCKET NUMBER: 50-304

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 1040 MWe

REACTOR AGE: 3.6 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Sargent & Lundy

OPERATORS: Commonwealth Edison Co.

LOCATION: 40 miles north of Chicago, Ill.

DURATION: N/A

PLANT OPERATING CONDITION: hot shutdown

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ______________________

DISCOVERY METHOD: operation event during testing

COMMENT: The existing test procedure required the use of dummy signals only when required. During the test, dummy signals were installed on all 31 instrument inputs even though none were needed.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 128569

Date: July 15, 1977

Title: Safety Relief Valve Fails to Reset at Brunswick 2

The failure sequence was:

1. A turbine trip occurred which resulted in a reactor scram.
2. HPCI and RCIC initiated and injected into the reactor causing a level increase and subsequent HPCI and RCIC trip on high water level.
3. Safety relief valves were opened three times to maintain the reactor pressure below 1050 psig. On the third opening the valve failed to close.
4. RCIC was started and injected to the reactor in an attempt to recover reactor vessel level. It was of insufficient capacity so APC1 was also started and the level was recovered.
5. (See next page)

Corrective action:

The grounded solenoid was rewired and retaped

Design purpose of failed system or component:

The safety relief valves are designed to control reactor pressure.

Unavailability of system per WASH 1400:* -

Unavailability of component per WASH 1400:* Relief valve failure to close: 10^{-2}/D

---

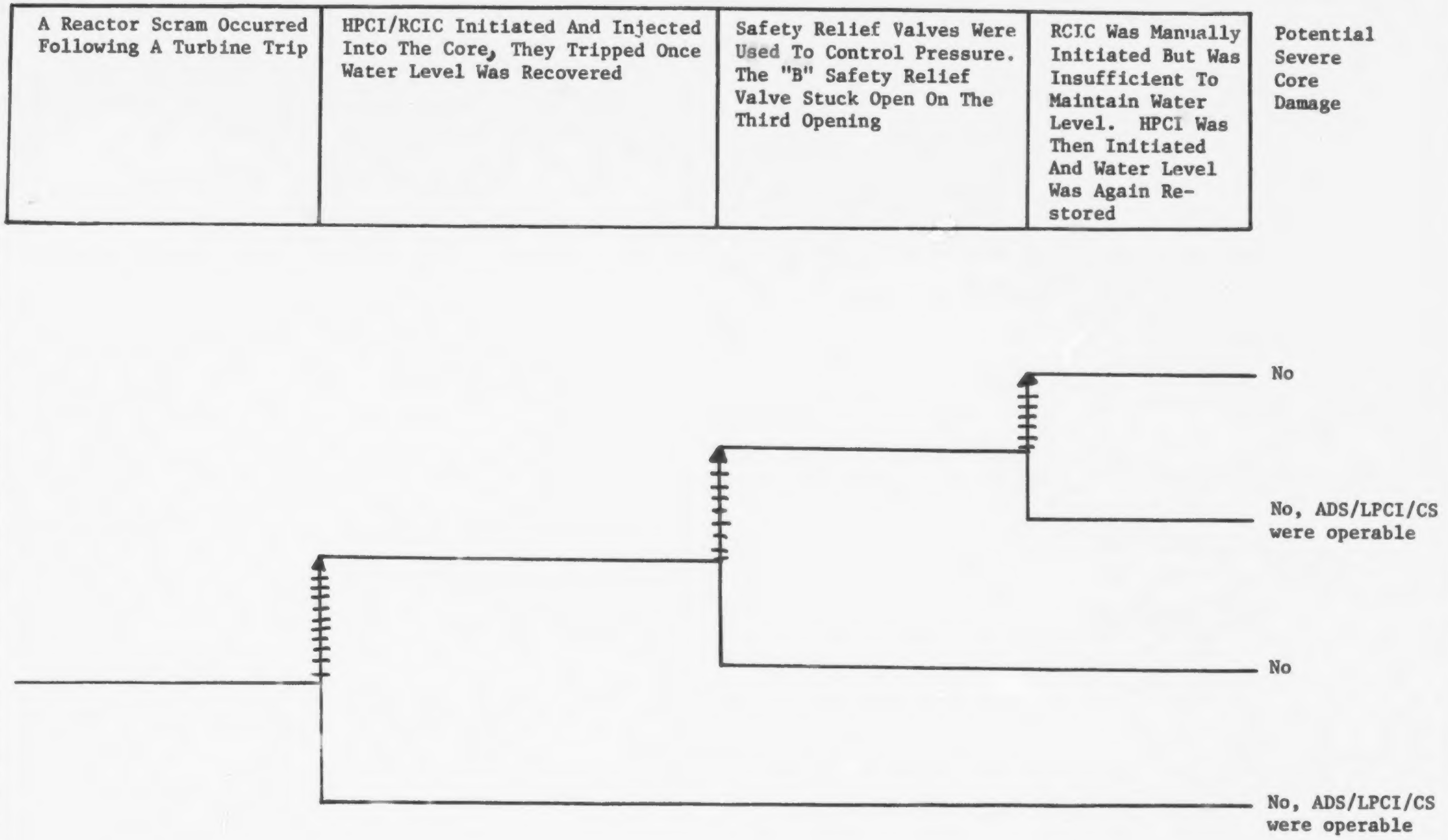
*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

[redacted]

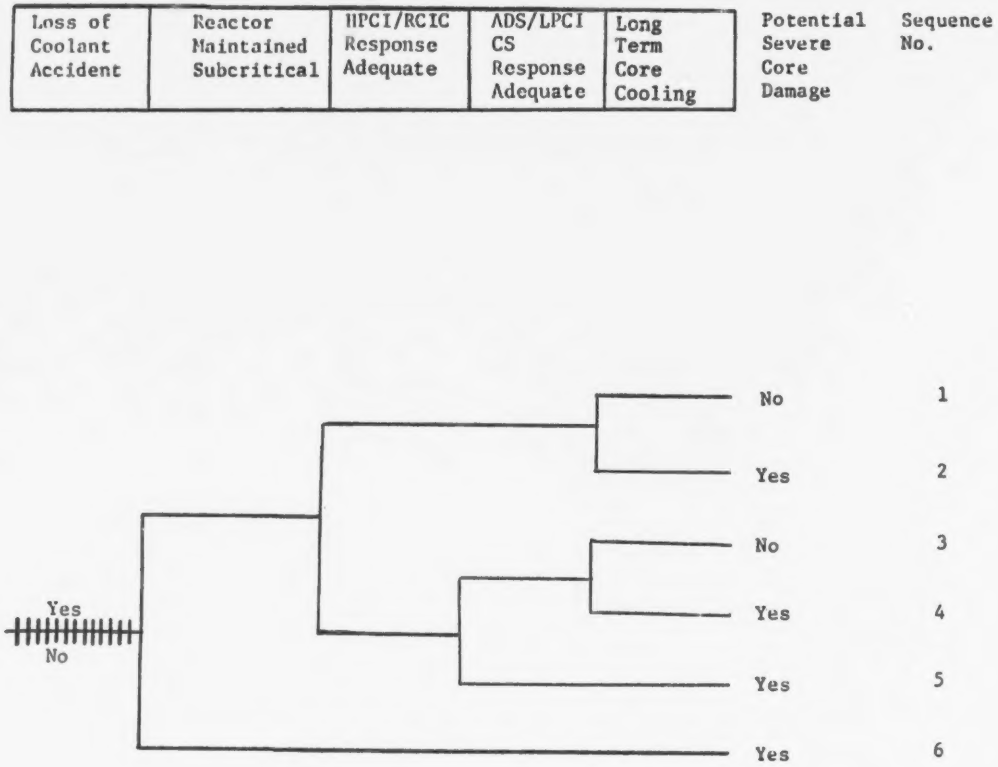
The Failure sequence was: (Continued)

5. The division IRHR system was placed in service to cool the torus.
6. The reactor pressure decreased from 1050 to 280 in 14 minutes corresponding to a cool down rate of 136F°/hr.
7. The valve failed to reset because the solenoid assembly was grounded.

BEST DOCUMENT AVAILABLE


A Reactor Scram Occurred Following A Turbine Trip
HPCI/RCIC Initiated And Injected Into The Core, They Tripped Once Water Level Was Recovered
Safety Relief Valves Were Used To Control Pressure. The "B" Safety Relief Valve Stuck Open On The Third Opening
RCIC Was Manually Initiated But Was Insufficient To Maintain Water Level. HPCI Was Then Initiated And Water Level Was Again Restored
Potential Severe Core Damage
No
No, ADS/LPCI/CS were operable
No
No, ADS/LPCI/CS were operable

NSIC 128569 — Actual Occurrence of Safety Relief Valve Fails to Reseat at Brunswick 2


Loss of Coolant Accident
Reactor Maintained Subcritical
HPCI/RCIC Response Adequate
ADS/LPCI CS Response Adequate
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
No
1
Yes
2
No
3
Yes
4
Yes
5
Yes
6

NSIC 128569 — Sequence of Interest for Safety Relief Valve Fails to Reset at Brunswick 2

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 128569

DATE OF LER: August 3, 1977

DATE OF EVENT: July 15, 1977

SYSTEM INVOLVED: Pressure Relief

COMPONENT INVOLVED:

CAUSE: Safety Relief Valve

SEQUENCE OF INTEREST: Loss of Coolant Accident

ACTUAL OCCURRENCE: Safety Relief Valve Fails to Reset.

REACTOR NAME: Brunswick 2

DOCKET NUMBER: 324

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 821 MWe

REACTOR AGE: 2.4 yr

VENDOR: G.E.

ARCHITECT-ENGINEERS: United Engineers

OPERATORS: Carolina Power and Light

LOCATION: Three miles N of Southport

DURATION: N/A

PLANT OPERATING CONDITION: Reactor Scram from 75% Power

SAFETY FEATURE TYPE OF FAILURE: [✓ (a)] inadequate performance; (b) failed to start; (c) made inoperable; (d) ______________

DISCOVERY METHOD: Operation Event

COMMENT:

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: **128906**

Date: **August 31, 1977**

Title: **No Break Power Panel De-energized at Cooper**

The failure sequence was:

1. **A blown fuse in the no break power panel (NBPP) caused it to de-energize. The automatic power backup source was out of service (due to a failed K10 contactor).**
2. **The NBPP supplies both the feedwater controller and the RCIC, thus a partial loss of feedwater flow and degraded operation of the RCIC resulted.**
3. **The reactor tripped on low water level.**
4. **The MSIVs closed and the HPCI and RCIC systems initiated on low-low water level.**
5. **Neither the RCIC nor the HPCI responded adequately. The RCIC did not go to full speed because of the loss of the NBPP. The HPCI, however, did not accelerate to full speed because the governor actuator was not functioning.**

(see attached page)

Corrective action:

1. **The blown fuse was replaced.**
2. **The contactor for the auxiliary power source was replaced and reconnected.**

Design purpose of failed system or component:

1. **The NBPP provides a continuous supply of power to vital loads.**
2. **The K10 contactor allows the NBPP to switch to an alternate source of power.**

Unavailability of system per WASH 1400:[*] **RCIC: 8.0×10^{-2}/D**
HPCI: 8.8×10^{-2}/D

Unavailability of component per WASH 1400:[*] **fuse 1×10^{-6}/hr, solid state devices 3×10^{-6}/hr**

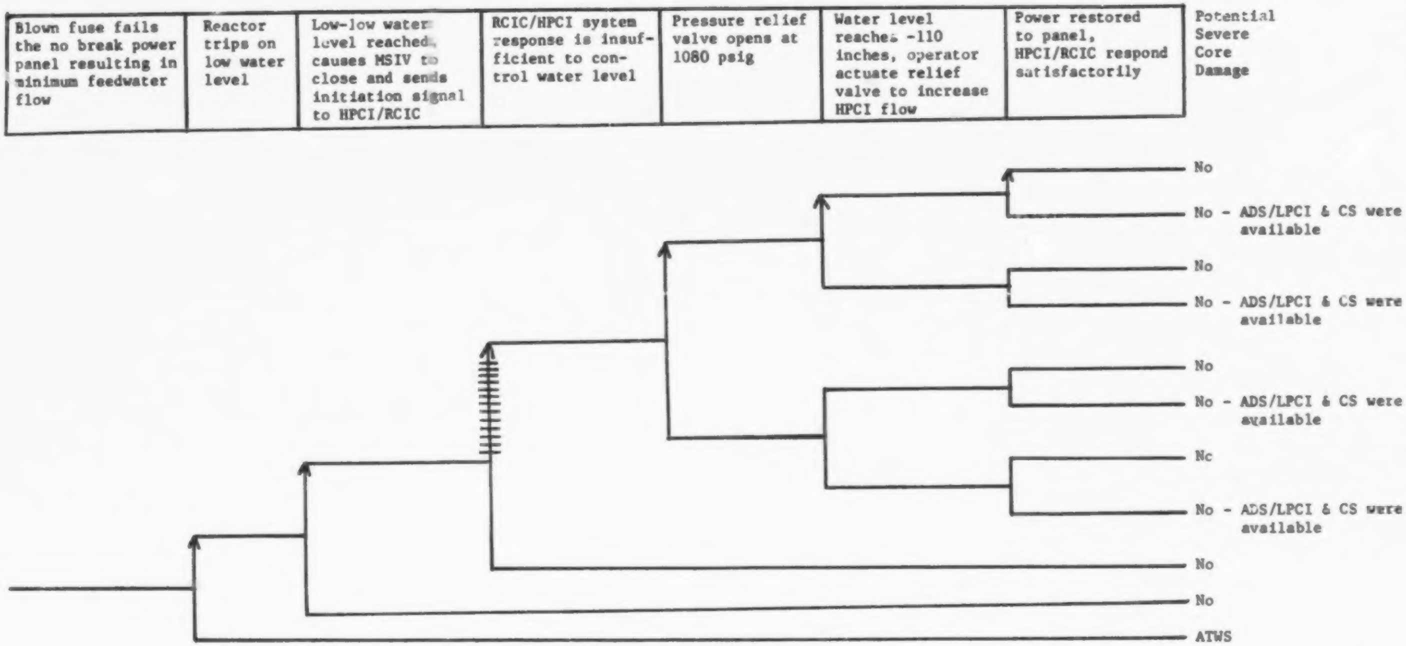
---

[*] Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

[redacted]

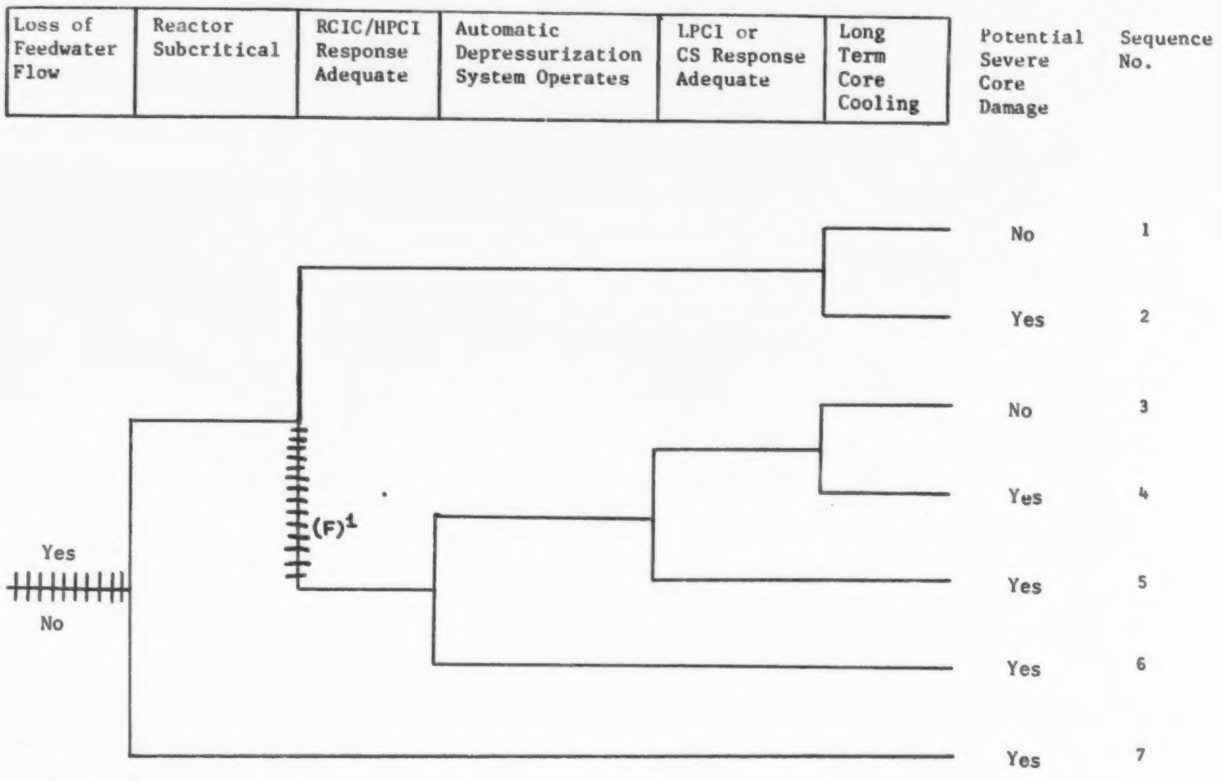
The failure sequence was: (continued)

6. A relief valve opened at 1080 psig for pressure control.
7. At -110" the operator opened another relief valve in order to increase HPCI flow. At that point power was restored and the RCIC pump ramped up to power and provided proper water level control.


Blown fuse fails the no break power panel resulting in minimum feedwater flow
Reactor trips on low water level
Low-low water level reached, causes MSIV to close and sends initiation signal to HPCI/RCIC
RCIC/HPCI system response is insufficient to control water level
Pressure relief valve opens at 1080 psig
Water level reaches -110 inches, operator actuate relief valve to increase HPCI flow
Power restored to panel, HPCI/RCIC respond satisfactorily
Potential Severe Core Damage
No
No - ADS/LPCI & CS were available
No
No - ADS/LPCI & CS were available
No
No - ADS/LPCI & CS were available
Nc
No - ADS/LPCI & CS were available
No
No
ATWS

NSIC 128906 – Actual Occurrence for No Break Power Panel De-energized at Cooper

383


Loss of Feedwater Flow
Reactor Subcritical
RCIC/HPCI Response Adequate
Automatic Depressurization System Operates
LPCI or CS Response Adequate
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
(F)1
No 1
Yes 2
No 3
Yes 4
Yes 5
Yes 6
Yes 7

NSIC 128906 – Sequence of Interest for No Break Power Panel De-energized at Cooper

[1] Success requires restoration of power to the NBPP.

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: **128906**

DATE OF LER: **September 30, 1977**

DATE OF EVENT: **August 31, 1977**

SYSTEM INVOLVED: **electric power, feedwater, RCIC, HPCI**

COMPONENT INVOLVED: **fuses and circuit breakers**

CAUSE: **a blown fuse**

SEQUENCE OF INTEREST: **loss of feedwater flow**

ACTUAL OCCURRENCE: **no break power panel de-energized at Cooper**

REACTOR NAME: **Cooper**

DOCKET NUMBER: **50-298**

REACTOR TYPE: **BWR**

DESIGN ELECTRICAL RATING: **778** MWe

REACTOR AGE: **3.5** yr

VENDOR: **GE**

ARCHITECT-ENGINEERS: **Burns and Roe**

OPERATORS: **Nebraska Public Power District**

LOCATION: **23 miles S of Nebraska City**

DURATION: N/A

PLANT OPERATING CONDITION: **92%**

SAFETY FEATURE TYPE OF FAILURE: [✓ (a)] inadequate performance; (b) failed to start; (c) made inoperable; (d) ____________________

DISCOVERY METHOD: **operational event**

COMMENT: -

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 128935

Date: August 29, 1977

Title: Plant 161 kV Feed Lost Momentarily at Ft. Calhoun 1

The failure sequence was:

1. A target for a backup relay at a substation was being reset after a trip caused by an electrical storm.
2. A defective reset mechanism on the target caused a momentary loss of the 161 kV line to the plant.
3. An automatic fast transfer to the 22 kV reserve feed did not occur due to the fact that such a transfer is initiated by the primary relaying at the same substation.
4. The loss of the 161 kV line and subsequent failure to transfer to the 22 kV line resulted in a loss of offsite power on the safety related buses at the plant.

(See next page)

Corrective action:

After shutdown was completed, the plant was back-fed from the 345 kV buses while the defective relay was repaired.

Design purpose of failed system or component:

The 161 kV offsite line is used to supply power to the safety related buses and to half of the higher voltage buses in the plant during plant operation.

Unavailability of system per WASH 1400:* Offsite power: 2×10^{-5}/hr

Unavailability of component per WASH 1400:* -

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

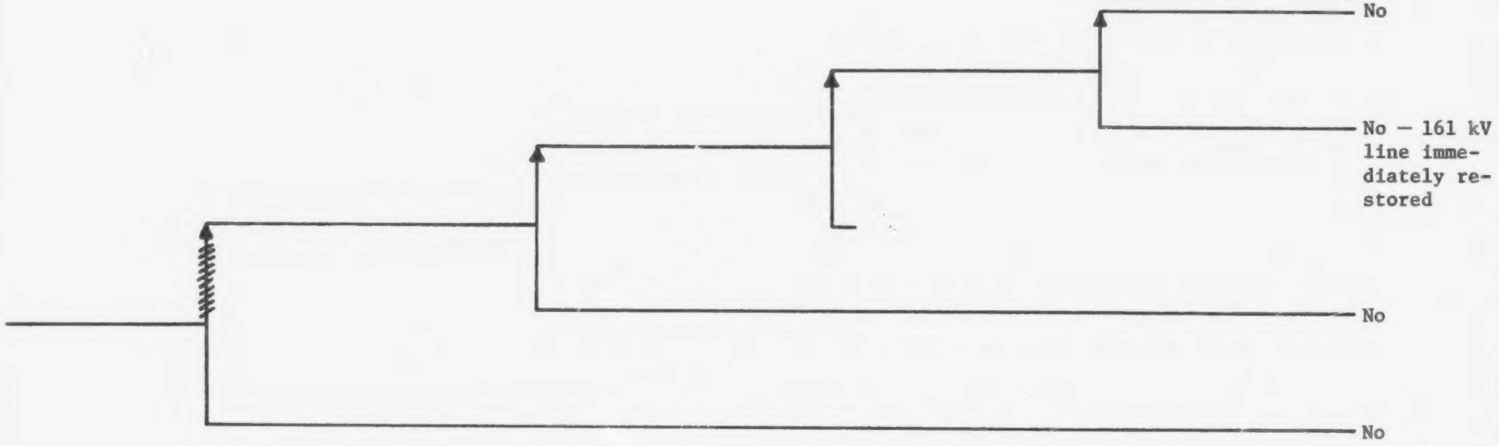
[redacted]

The failure sequence was: (continued)

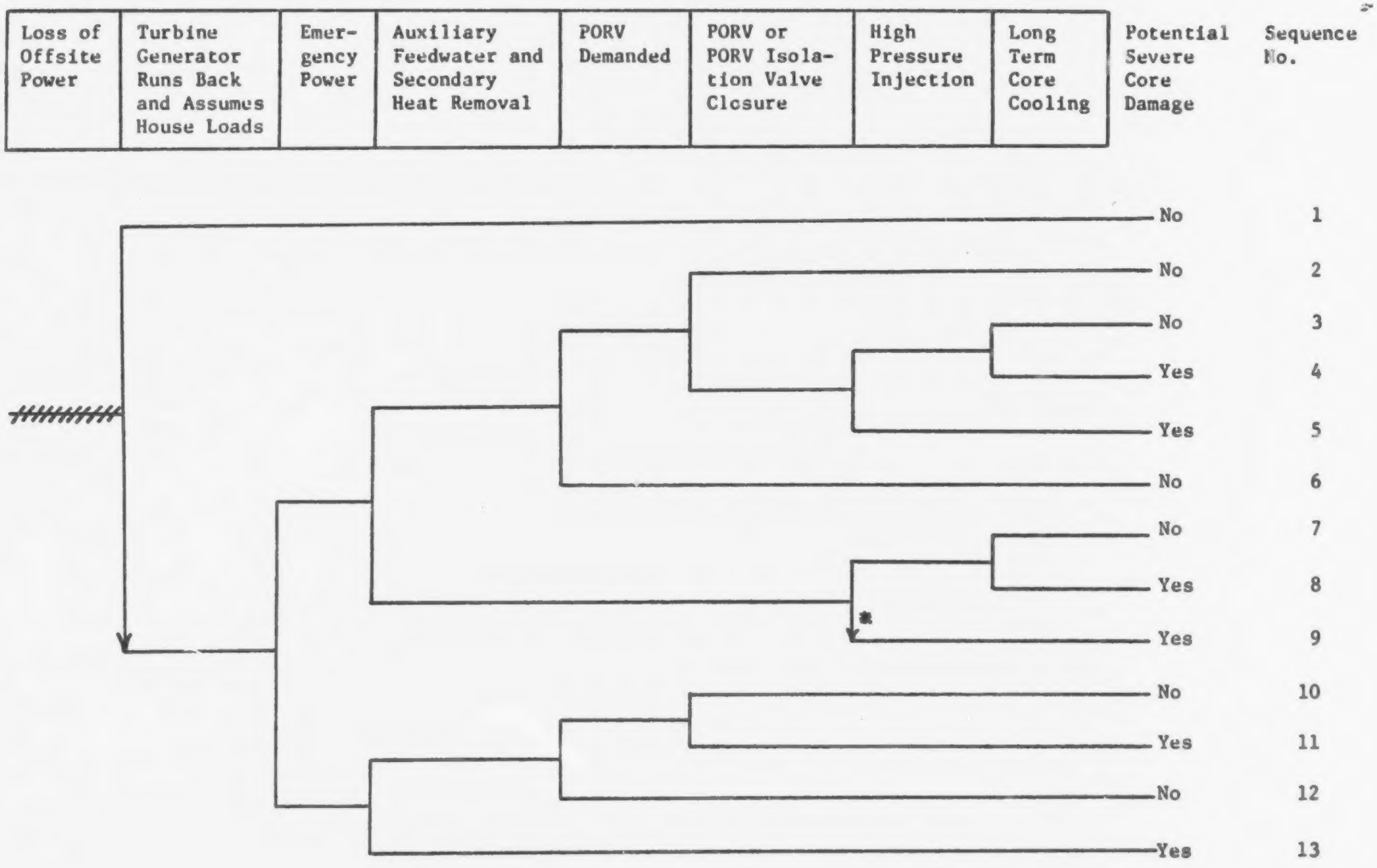
5. This loss of voltage resulted in the tripping of two reactor coolant pumps and a subsequent reactor trip on low coolant flow.
6. The diesel generators automatically started and supplied power to the safety-related buses.
7. The substation was restored immediately after tripping.

BEST DOCUMENT AVAILABLE

Reactor at 98% Power and Back-up Relay Target at 161 kV Substation Being Reset	Defective Reset Mechanism on Backup Relay Results in Loss of 161 kV Lines to Plant	Fast Transfer to 22 kV Reserve Feed Does Not Occur Due to Fact That Such Transfers are Initiated by Primary Relaying at 161 kV Substation	Loss of 161 kV Power Results in Trip of Two Reactor Coolant Pumps and Reactor Trip on Low Coolant Flow	Diesel Generators Start and Provide Power to Safety-Related Buses	Potential Severe Core Damage



NSIC 128935 — Actual Occurrence for Plant 161 kV Feed Lost Momentarily at Ft. Calhoun 1


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
*
Yes 9
No 10
Yes 11
No 12
Yes 13

NSIC 128935 — Sequence of Interest for Plant 161 kV Feed Lost Momentarily at Ft. Calhoun 1

*Not included in mitigation procedures.

389

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 128935

DATE OF LER: August 29, 1977

DATE OF EVENT: August 22, 1977

SYSTEM INVOLVED: Offsite power

COMPONENT INVOLVED: backup relay target mechanism

CAUSE: Failure cause not specified

SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE: Loss of offsite power

REACTOR NAME: Fort Calhoun 1

DOCKET NUMBER: 50-285

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 457 MWe

REACTOR AGE: 4.0 yr

VENDOR: Combustion Engineering

ARCHITECT-ENGINEERS: Gibbs & Hill

OPERATORS: Omaha Public Power District

LOCATION: 19 mi north of Omaha, Nebraska

DURATION: N/A

PLANT OPERATING CONDITION: 98% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c) made inoperable]; (d) ____________________

DISCOVERY METHOD: operational event

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 130111

Date: September 28, 1977

Title: Blackout Condition Occurs After Reactor Trip at Cook 1

The failure sequence was:

1. With the reactor at 100% power and during an extreme thunderstorm, a lightning strike on a transmission circuit resulted in the opening of transformer breakers and the de-energization of the "normal reserve" transformer.
2. A subsequent lightning strike resulted in a momentary 36% voltage dip that was sensed by the reactor coolant pump bus undervoltage relays, which prematurely initiated a reactor trip and subsequent turbine and generator trips. This resulted in a loss of station electric power.
3. The two diesel generators started and provided power to essential loads.
4. One hour and 13 minutes after reactor trip, the decision was made to transfer the emergency buses to the "emergency reserve" offsite source (which had been available throughout the incident). Three of the four buses successfully transferred, the fourth bus did not transfer (reason unspecified) and was supplied power from its diesel generators.

Corrective action:

1. The reactor coolant pump bus undervoltage relays were reset to their correct delay times.
2. The reason for the failure of one emergency bus to transfer to the "emergency reserve" source was not specified.

Design purpose of failed system or component:

Offsite power provides the preferred source of power to plant loads when the unit generator is not operating.

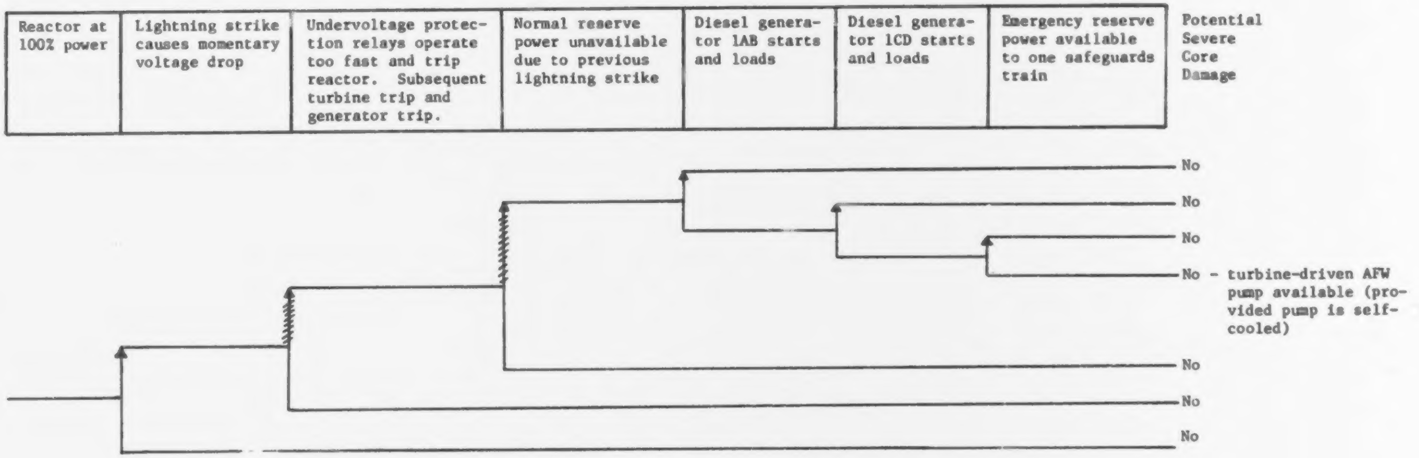
Unavailability of system per WASH 1400:* Offsite power: 2×10^{-5}/hr (following trip: 1×10^{-3}/D)

Unavailability of component per WASH 1400:* —

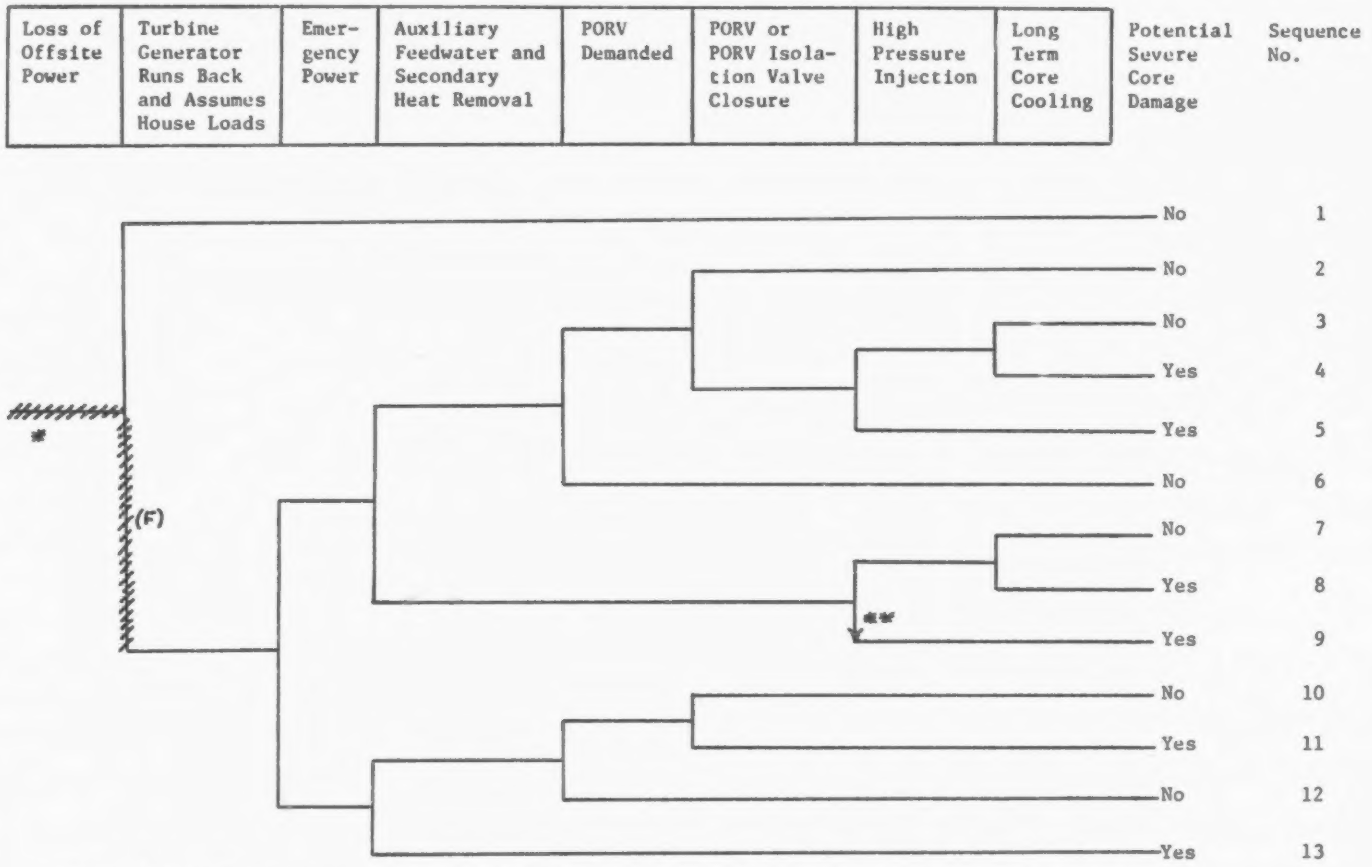
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

BEST DOCUMENT AVAILABLE


Reactor at 100% power
Lightning strike causes momentary voltage drop
Undervoltage protection relays operate too fast and trip reactor. Subsequent turbine trip and generator trip.
Normal reserve power unavailable due to previous lightning strike
Diesel generator 1AB starts and loads
Diesel generator 1CD starts and loads
Emergency reserve power available to one safeguards train
Potential Severe Core Damage
No
No
No
No - turbine-driven AFW pump available (provided pump is self-cooled)
No
No
No

NSIC 130111 — Actual Occurrence for "Blackout" Condition Occurs After Reactor Trip at Cook 1


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
*
(F)
**
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
Yes 9
No 10
Yes 11
No 12
Yes 13

NSIC 130111 — Sequence of Interest for "Blackout" Condition Occurs After Reactor Trip at Cook 1

*success requires manual transfer to an emergency reserve power source.

**not included in mitigation procedures.

BEST DOCUMENT AVAILABLE

393

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 130111

DATE OF LER: September 28, 1977

DATE OF EVENT: September 1, 1977

SYSTEM INVOLVED: Offsite power, on-site power

COMPONENT INVOLVED: reserve power transformer breakers
reactor coolant pump bus undervoltage relays

CAUSE: The breakers opened to clear a fault, the undervoltage relays operated prematurely.

SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE: Reactor trip and loss of offsite power

REACTOR NAME: Cook 1

DOCKET NUMBER: 50-315

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 1054 MWe

REACTOR AGE: 2.6 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: American Electric Power

OPERATORS: Indiana & Michigan Power Co. (American Electric Power)

LOCATION: 11 mi. South of Benton Harbor, Michigan

DURATION: N/A

PLANT OPERATING CONDITION: 100% power

SAFETY FEATURE TYPE OF FAILURE: [✓ (a)] inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ______________________

DISCOVERY METHOD: Operational event

COMMENT: –

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 130119

Date: September 24, 1977

Title: A Complete Loss of Offsite Power Occurs at Palisades

The failure sequence was:

1. With the reactor at 100% power during an electrical storm, the switchyard "R" bus de-energized and caused a complete loss of offsite power and loss of main condenser cooling.
2. The turbine tripped on high condenser vacuum, and effected reactor and generator trips.
3. Both diesel generators started and supplied electric power to safety-related loads.

Corrective action:

none

Design purpose of failed system or component:

Off-site power provides the preferred source of electric power to plant equipment when the unit generator is not in operation. The condenser circulating water pumps are normally powered from the offsite power source.

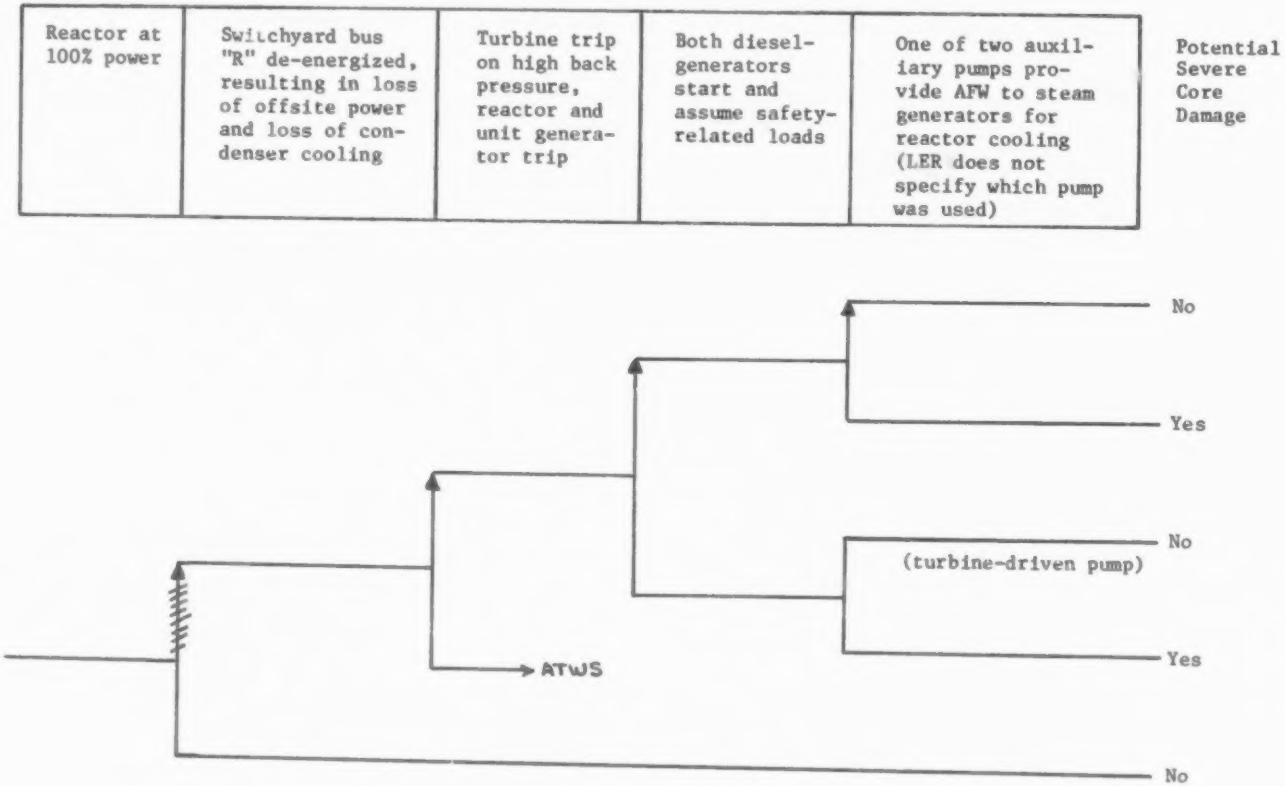
Unavailability of system per WASH 1400:* loss of offsite power: 2×10^{-5}/hr

Unavailability of component per WASH 1400:* —

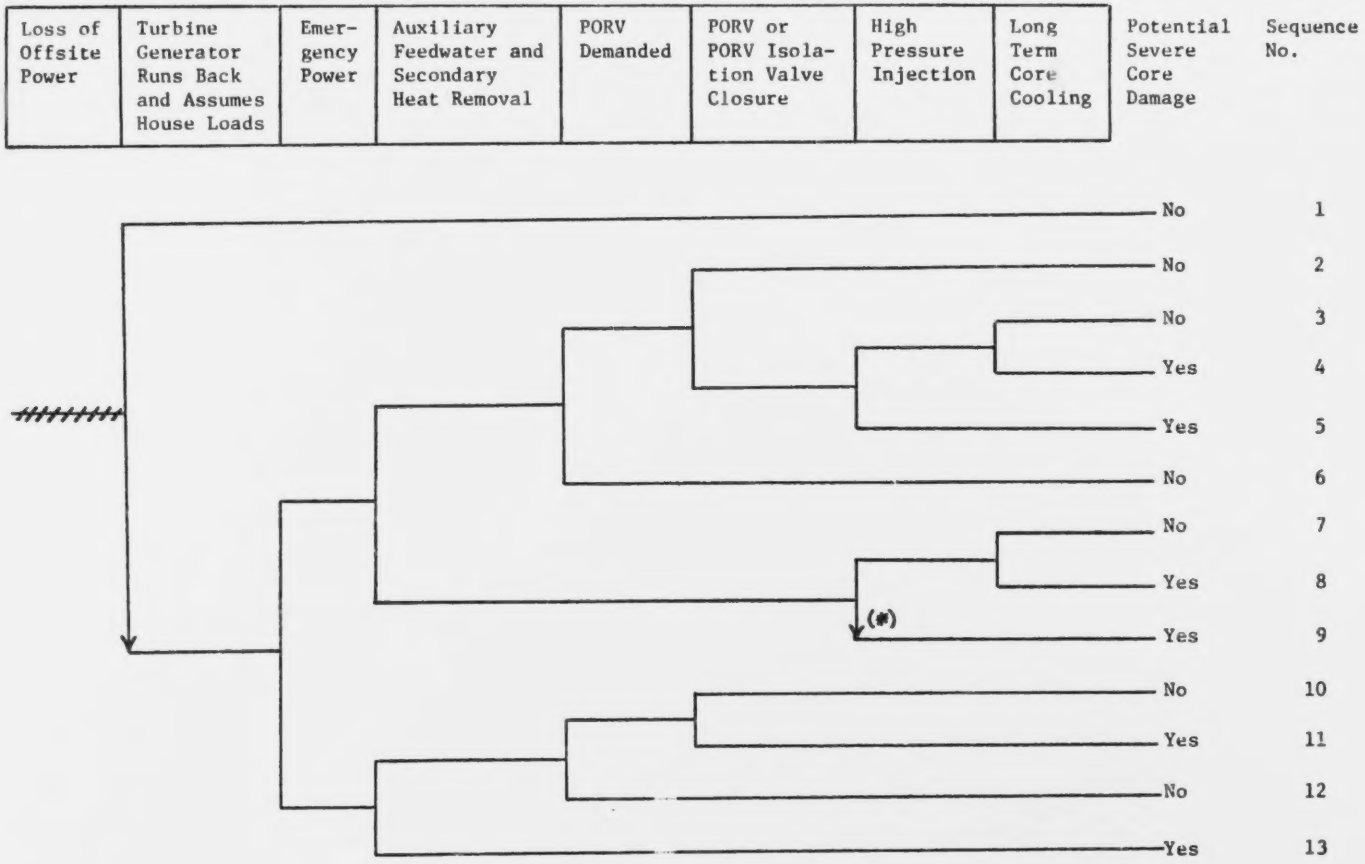
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

BEST DOCUMENT AVAILABLE


Reactor at 100% power
Switchyard bus "R" de-energized, resulting in loss of offsite power and loss of condenser cooling
Turbine trip on high back pressure, reactor and unit generator trip
Both diesel-generators start and assume safety-related loads
One of two auxiliary pumps provide AFW to steam generators for reactor cooling (LER does not specify which pump was used)
Potential Severe Core Damage
No
Yes
No
(turbine-driven pump)
Yes
ATWS
No

NSIC 130119 — Actual Occurrence for a Complete Loss of Offsite Power at Palisades


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
(*)
Yes 9
No 10
Yes 11
No 12
Yes 13

NSIC 130119 — Sequence of Interest for a Complete Loss of Offsite Power at Palisades
*Not included in mitigation procedures.

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 130119

DATE OF LER: October 18, 1977

DATE OF EVENT: September 24, 1977

SYSTEM INVOLVED: offsite power

COMPONENT INVOLVED: switchyard "R" bus

CAUSE: bus trip from unknown causes

SEQUENCE OF INTEREST: loss of offsite power

ACTUAL OCCURRENCE: reactor trip due to loss of offsite power

REACTOR NAME: Palisades

DOCKET NUMBER: 50-255

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 805 MWe

REACTOR AGE: 6.5 yr

VENDOR: Combustion Engineering

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Consumers Power Co.

LOCATION: 5 miles south of South Haven, Mich.

DURATION: N/A

PLANT OPERATING CONDITION: 100% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ________________

DISCOVERY METHOD: during operation

COMMENT: –

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 130788

Date: October 7, 1977

Title: Half-trip of SFRCS Causes RCS Pressure Rise at Davis-Besse 1

The failure sequence was:

1. A spurious half-trip of the SFRCS occurred, which initiated closure of a feedwater valve, resulting in loss of water to one steam generator.
2. The reduction in heat removal resulted in operation of the pressurizer relief valve, which failed open after 9 open/close cycles.
3. The operator manually tripped the reactor; within six minutes the RCS was at saturation pressure/temperature.
4. The pressurizer level went to maximum.
5. Approximately 25 minutes later the operator realized the relief valve was stuck open and closed the block valve, terminating the transient.

Corrective action:

1. The SFRCS was monitored to detect spurious signals.
2. Plans were also being developed to add additional SFRCS alarm windows and seal in alarm conditions.
3. A missing PORV control circuit relay was installed.

Design purpose of failed system or component:

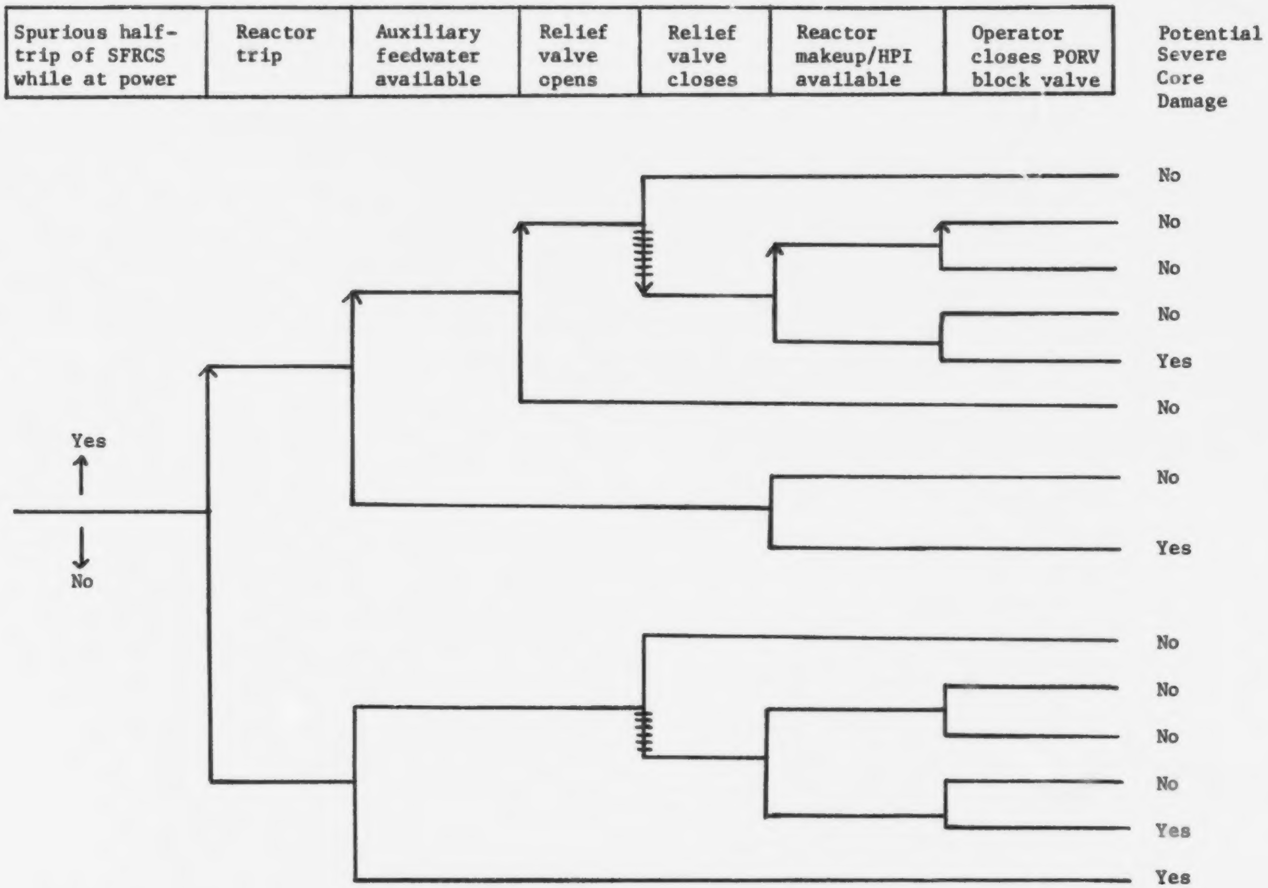
1. Steam generator isolation during a steam line break.

Unavailability of system per WASH 1400:* SFRCS — type systems were not considered in WASH-1400.

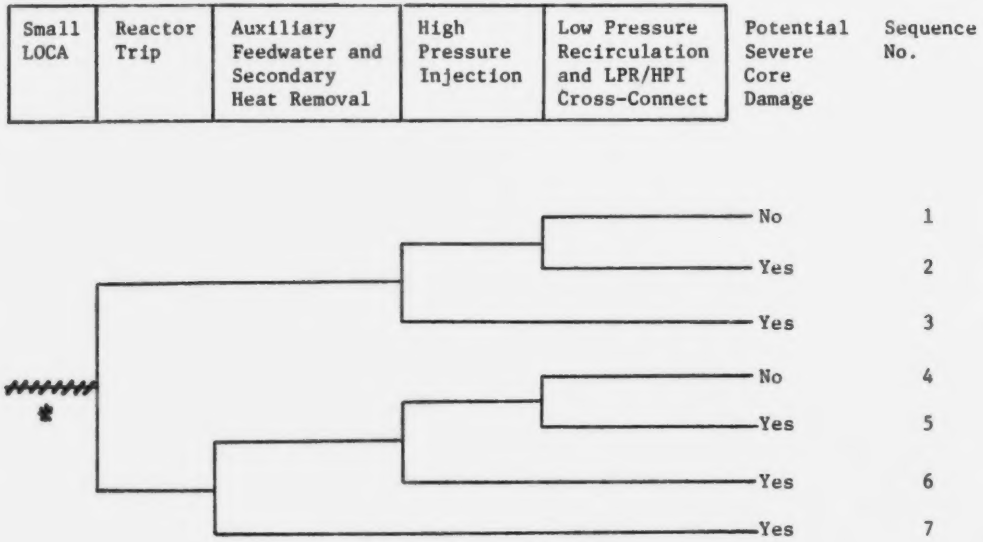
Unavailability of component per WASH 1400:*

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Spurious half-trip of SFRCS while at power
Reactor trip
Auxiliary feedwater available
Relief valve opens
Relief valve closes
Reactor makeup/HPI available
Operator closes PORV block valve
Potential Severe Core Damage
Yes
No
No
No
No
No
Yes
No
No
Yes
No
No
No
No
Yes
Yes

NSIC 130788 — Actual Occurrence for Spurious Half-Trip fo SFRCS Causes RCS Pressure Rise at Davis-Besse 1


Small LOCA
Reactor Trip
Auxiliary Feedwater and Secondary Heat Removal
High Pressure Injection
Low Pressure Recirculation and LPR/HPI Cross-Connect
Potential Severe Core Damage
Sequence No.
*
No 1
Yes 2
Yes 3
No 4
Yes 5
Yes 6
Yes 7

NSIC 130788 — Sequence of Interest for Spurious Half-Trip of SFRCS Causes RCS Pressure Rise at Davis-Besse 1

*failure requires operator error in failing to close PORV block valve.

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 130788

DATE OF LER: October 7, 1977

DATE OF EVENT: September 24, 1977

SYSTEM INVOLVED: Pressure relief of RC system

COMPONENT INVOLVED: Valve

CAUSE: Close relay missing from relief valve control circuit (PORV failure).

SEQUENCE OF INTEREST: Stuck open PORV — small LOCA.

ACTUAL OCCURRENCE: Spurious 1/2 trip of steam and feedwater rupture control system.

REACTOR NAME: Davis-Besse 1

DOCKET NUMBER: 50-346

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 906 MWe

REACTOR AGE: 0.5 yr

VENDOR: B&W

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Toledo Edison Co.

LOCATION: 21 miles east of Toledo, Ohio

DURATION: N/A

PLANT OPERATING CONDITION: 9% power

SAFETY FEATURE TYPE OF FAILURE: [✓ (a)] inadequate performance; (b) failed to start; (c) made inoperable; (d) ____________

DISCOVERY METHOD: Operational error.

COMMENT: No cause found for the 1/2 trip valve stuck open because pilot valve stem failed.

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 130889

Date: November 15, 1977

Title: Emergency Service Water Valves Found Flooded at Surry 2

The failure sequence was:

1. During a refueling outage, drain **valves** associated with the service water lines to the recirculation heat exchangers were left open.
2. When one of the recirculation service water isolation valves was opened, the open drain valves caused flooding of the valve pit, submerging all four recirculation heat exchanger service water isolation valve motor operators. These valves must operate during a LOCA.

Corrective action:

1. The valve pit was drained and the drain valves shut.
2. The valve motor operators were repaired.
3. Personnel was stationed at the valve pit to open the service water valves if required.

Design purpose of failed system or component:

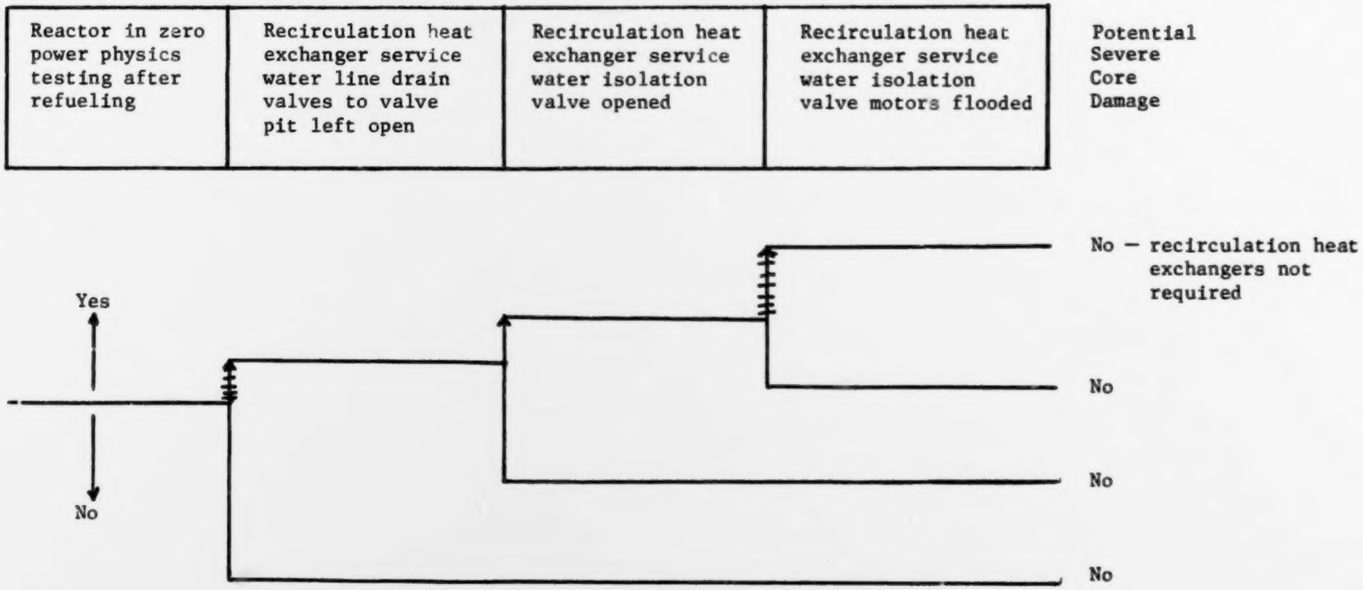
The valves provide service water flow to the recirculation heat exchangers for recirculation cooling following a LOCA. The containment spray recirculation is the only cooling provided during the recirculation phase.

Unavailability of system per WASH 1400:* Containment Spray Recirculation System: 1×10^{-4}/D (first 24 hours)

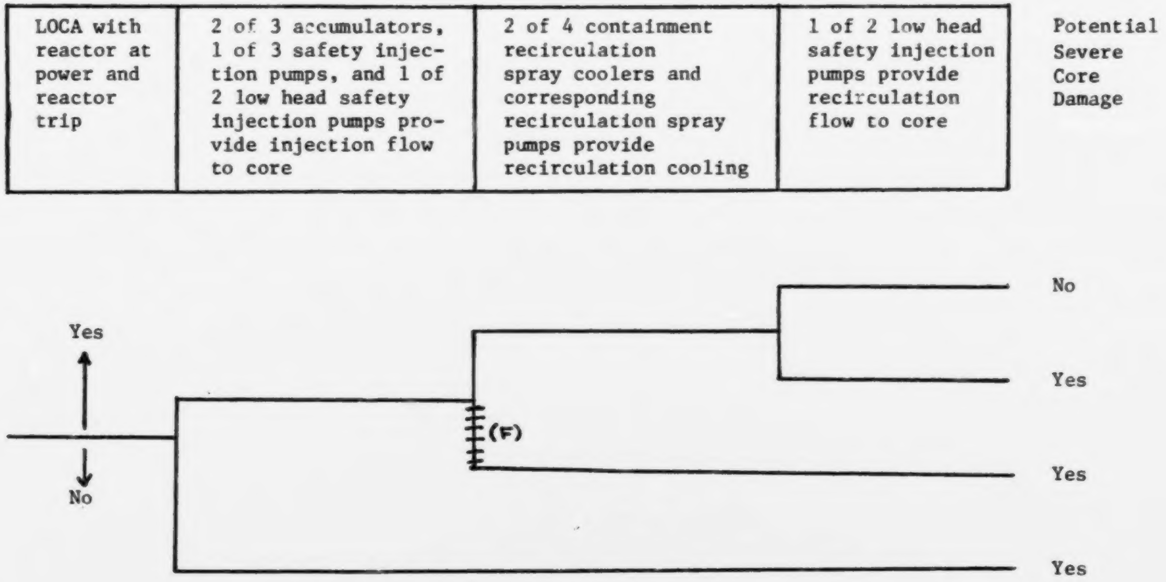
Unavailability of component per WASH 1400:* –

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor in zero power physics testing after refueling
Recirculation heat exchanger service water line drain valves to valve pit left open
Recirculation heat exchanger service water isolation valve opened
Recirculation heat exchanger service water isolation valve motors flooded
Potential Severe Core Damage
Yes
No
No — recirculation heat exchangers not required
No
No
No

NSIC 130889 — Actual Occurrence for Emergency Service Water Valves Found Flooded at Surry 2


LOCA with reactor at power and reactor trip
2 of 3 accumulators, 1 of 3 safety injection pumps, and 1 of 2 low head safety injection pumps provide injection flow to core
2 of 4 containment recirculation spray coolers and corresponding recirculation spray pumps provide recirculation cooling
1 of 2 low head safety injection pumps provide recirculation flow to core
Potential Severe Core Damage
Yes
No
(F)
No
Yes
Yes
Yes

NSIC 130889 — Sequence of Interest for Emergency Service Water Valves Found Flooded at Surry 2

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 130889

DATE OF LER: November 15, 1977

DATE OF EVENT: October 11, 1977

SYSTEM INVOLVED: Service Water System

COMPONENT INVOLVED: Recirculation Heat Exchanger Service Water Isolation Valves

CAUSE: Valve Motor Operators Flooded due to personnel error, human error

SEQUENCE OF INTEREST: LOCA

ACTUAL OCCURRENCE: Flooding of Valve Pit During Zero Power Physics Testing

REACTOR NAME: Surry 2

DOCKET NUMBER: 50-281

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 822 MWe

REACTOR AGE: 4.9 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Stone & Webster

OPERATORS: Virginia Electric & Power Co.

LOCATION: 17 miles NW of Newport News, Va.

DURATION: 360(a) hours

PLANT OPERATING CONDITION: Zero Power Physics Testing

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c) made inoperable]; (d) ______

DISCOVERY METHOD: During maintenance

COMMENT: –

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 131225

Date: August 28, 1977

Title: Two Diesel Generators Fail to Start at Peach Bottom 2

The failure sequence was:

1. During loss of offsite power test diesel E1 failed to start and diesel E4 failed to run after starting.
2. The problem with E1 was traced to jacket temperature alarm which had tripped in high jacket temperature during previous actuation. The service water pump supplying the jacket had tripped on thermal overload. The problem had been detected prior to the test, however, due to human oversight, it was not corrected.
3. An investigation showed that diesel E-4 tripped on overspeed. The overspeed mechanism had been installed improperly after maintenance during which the camshaft was replaced. The necessary shims for the overspeed mechanism were not replaced. This resulted in improper operation of the overspeed mechanism.

Corrective action:

1. The operators were reminded through a letter about the requirement to reset the diesel trips and alarms, after investigating the alarm.
2. The shims were installed in the overspeed mechanism.

Design purpose of failed system or component:

1. Emergency Power provides power when offsite power is unobtainable.
2. The overspeed mechanisms trip the diesel when it overspeeds.
3. The jacket alarm prevents the diesel from overheating.

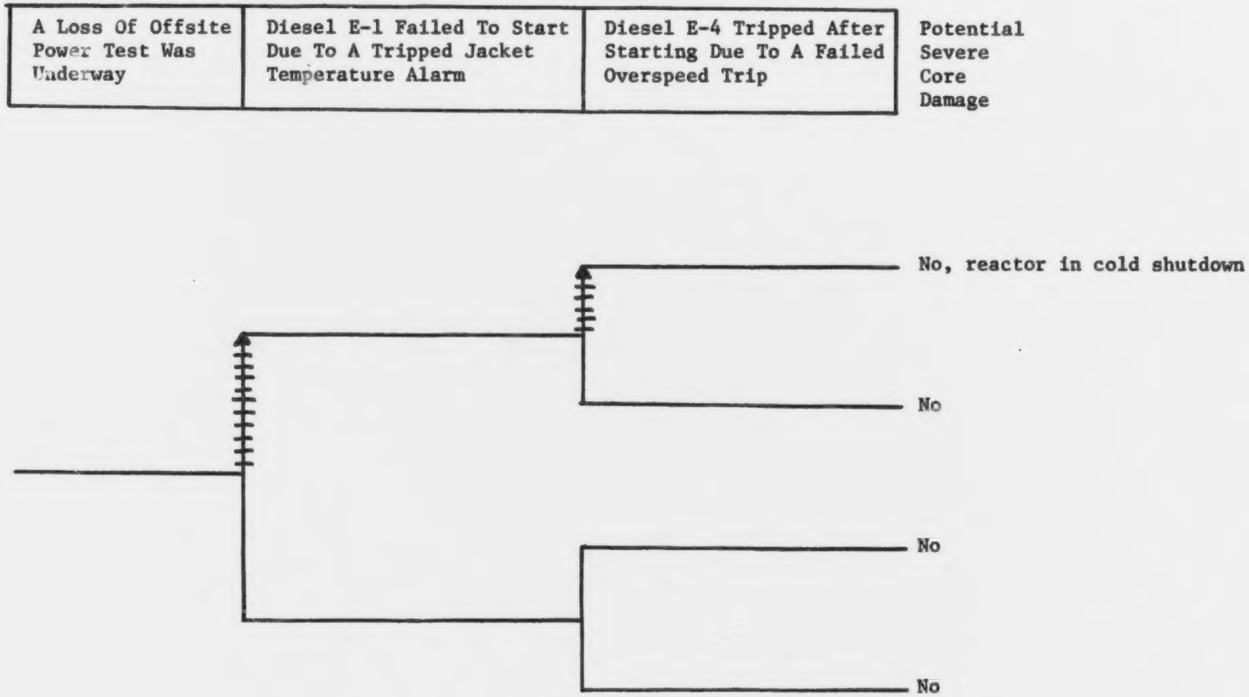
Unavailability of system per WASH 1400: *Emergency Power: 10^{-3}/D

Unavailability of component per WASH 1400: *Diesel Engine Failure to start: $3{:}10^{-2}$/D

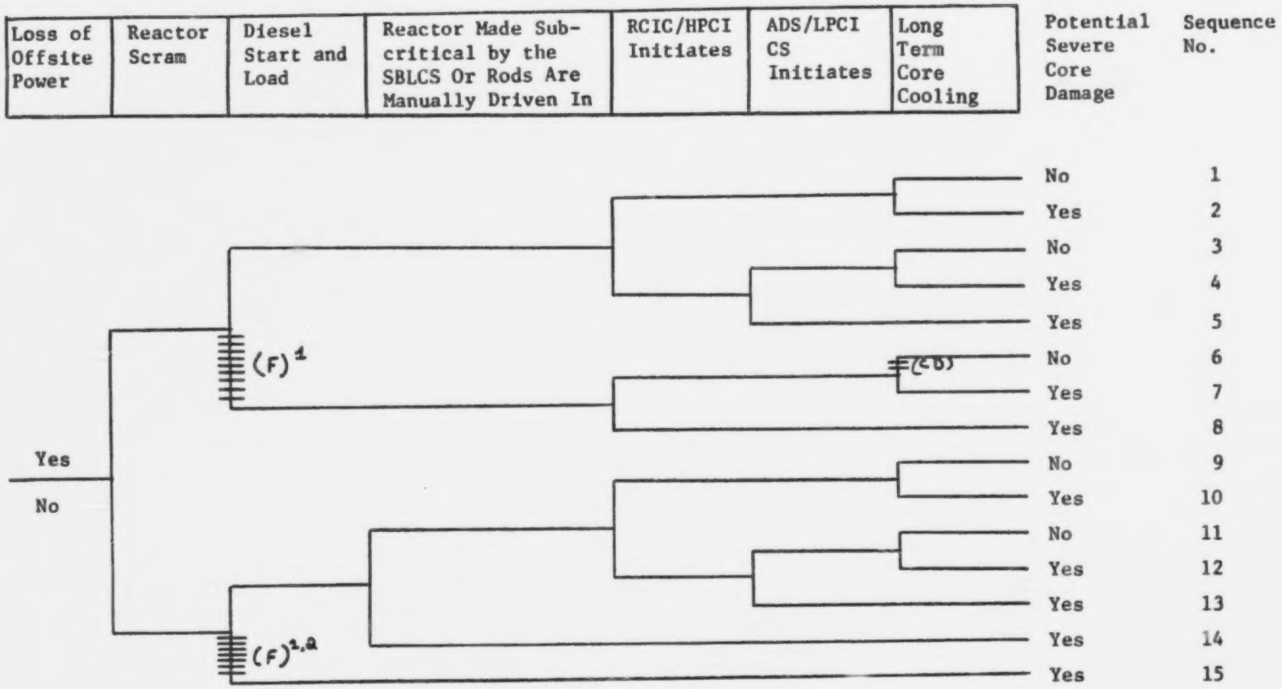
Diesel Engine Failure to run: 3.10^{-4}/Hr

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


A Loss Of Offsite Power Test Was Underway
Diesel E-1 Failed To Start Due To A Tripped Jacket Temperature Alarm
Diesel E-4 Tripped After Starting Due To A Failed Overspeed Trip
Potential Severe Core Damage
No, reactor in cold shutdown
No
No
No

NSIC 131225 — Actual Occurrence of Two Diesel Generators Fail at Peach Bottom 2


Loss of Offsite Power
Reactor Scram
Diesel Start and Load
Reactor Made Sub-critical by the SBLCS Or Rods Are Manually Driven In
RCIC/HPCI Initiates
ADS/LPCI CS Initiates
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
(F)1
(CD)
(F)1,2
No 1
Yes 2
No 3
Yes 4
Yes 5
No 6
Yes 7
Yes 8
No 9
Yes 10
No 11
Yes 12
Yes 13
Yes 14
Yes 15

NSIC 131225 — Sequence of Interest for Two Diesel Generators Fail to Start at Peach Bottom 2

[1] Resetting the jacket temperature alarm would result in a success.

[2] Proper loading of the operating diesels would allow success of alternate shutdown procedures or systems.

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 131225

DATE OF LER: September 9, 1977

DATE OF EVENT: August 26, 1977

SYSTEM INVOLVED: Emergency Electric Power

COMPONENT INVOLVED: Diesel Generators

CAUSE: Human oversight, Improper Maintenance & Procedures.

SEQUENCE OF INTEREST: Loss of Offsite Power.

ACTUAL OCCURRENCE: Two Diesels Fail to Start

REACTOR NAME: Peach Bottom 2

DOCKET NUMBER: 50-277

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 1065 MWe

REACTOR AGE: 3.9 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Philadelphia Electric Company

LOCATION: 19 miles S of Lancaster

DURATION: 360(a) hours

PLANT OPERATING CONDITION: 0% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________

DISCOVERY METHOD: Test

COMMENT:

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 132155

Date: November 29, 1977

Title: Diesel Generator Trips During Testing at Dresden 3

The failure sequence was:

1. Surveillance testing of diesel generator #2/3 was being performed when it tripped after 10 minute of operation. An investigation revealed that its cooling water pump had tripped and was inoperable. A mixture of oil and water had flooded the stator winding enclosure. The stator and associated bearings were ruined. The mixture leaked in through the conduit connected to the pump. No explanation was given for this.
2. An operability test of diesel generators 2 and 3 was immediately performed. Diesel generator 3 started but tripped 30 minutes after loading. A capacitor in the speed sensing circuit had shorted out.
3. An orderly shutdown of unit 3 was initiated.

Corrective action:

1. The damaged cooling water pump was replaced.
2. The shorted capacitor was replaced.

Design purpose of failed system or component:

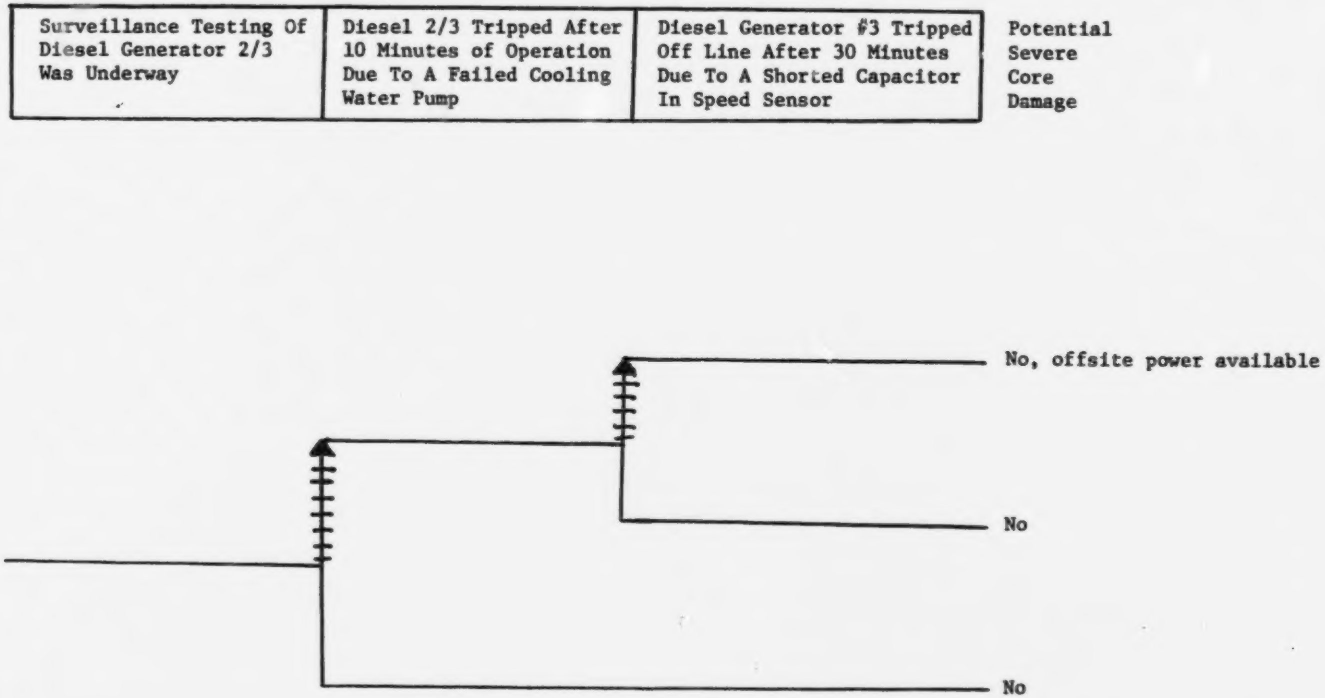
Emergency power provides a.c. power when offsite sources are unavailable.

Unavailability of system per WASH 1400:* Emergency power: 10^{-2}/D

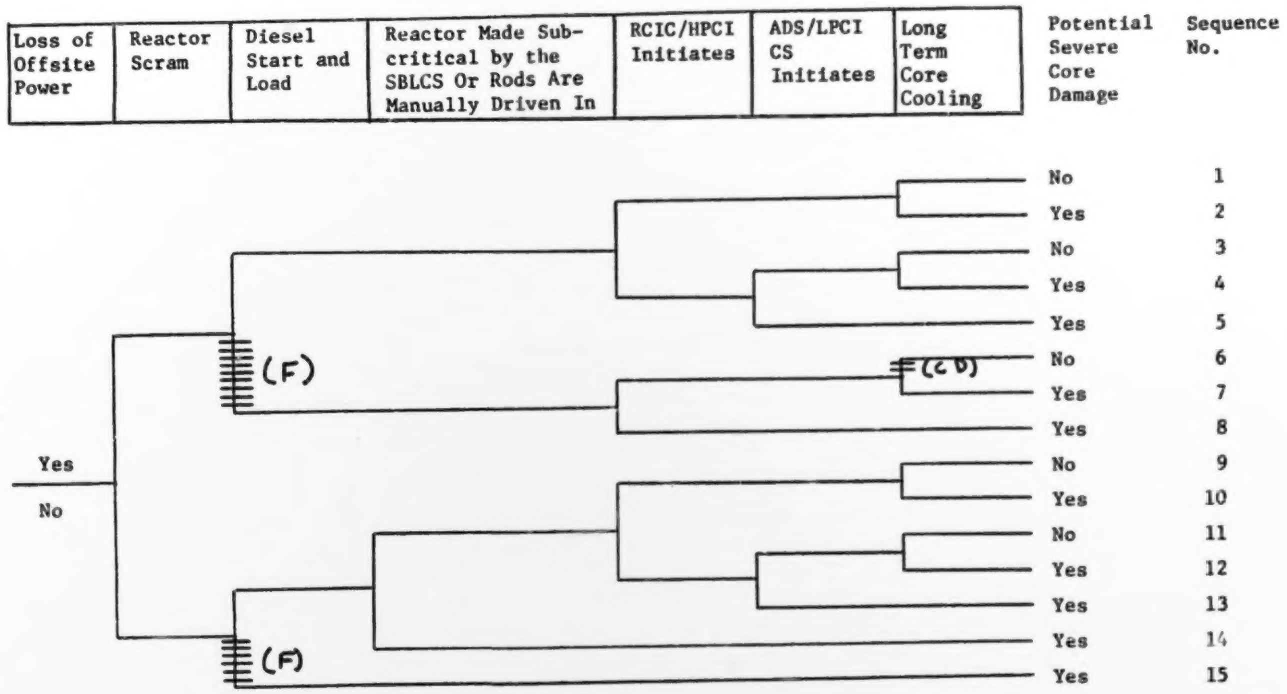
Unavailability of component per WASH 1400:* Diesel Engine failure to run:
3×10^{-4}/hr

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Surveillance Testing Of Diesel Generator 2/3 Was Underway
Diesel 2/3 Tripped After 10 Minutes of Operation Due To A Failed Cooling Water Pump
Diesel Generator #3 Tripped Off Line After 30 Minutes Due To A Shorted Capacitor In Speed Sensor
Potential Severe Core Damage
No, offsite power available
No
No

NSIC 132155 — Actual Occurrence of a Diesel Generator Trips During Testing of Dresden 3


Loss of Offsite Power
Reactor Scram
Diesel Start and Load
Reactor Made Subcritical by the SBLCS Or Rods Are Manually Driven In
RCIC/HPCI Initiates
ADS/LPCI CS Initiates
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
(F)
(CD)
(F)
No 1
Yes 2
No 3
Yes 4
Yes 5
No 6
Yes 7
Yes 8
No 9
Yes 10
No 11
Yes 12
Yes 13
Yes 14
Yes 15

NSIC 132155 — Sequence of Interest of a Diesel Generator Trips During Testing at Dresden 3

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 132155

DATE OF LER: December 13, 1977

DATE OF EVENT: November 29, 1977

SYSTEM INVOLVED: Emergency Power

COMPONENT INVOLVED: Diesel Generators

CAUSE: mechanical and electrical failures

SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE: Diesel Generator Trips During Testing at Dresden 3

REACTOR NAME: Dresden 3

DOCKET NUMBER: 50-249

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 794 MWe

REACTOR AGE: 6.8 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: Sargent & Lundy

OPERATORS: Commonwealth Edison

LOCATION: Nine miles E of Morris, Ill.

DURATION: 360(a) hours

PLANT OPERATING CONDITION: 75% power

SAFETY FEATURE TYPE OF FAILURE: [✓ (a)] inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________________

DISCOVERY METHOD: testing

COMMENT:

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 132927

Date: December 16, 1977

Title: Both 13.8 kV Buses De-energized at Davis-Besse 1

The failure sequence was:

1. Immediately following a reactor/turbine trip, the operator opened the main generator breakers and attempted a manual transfer to the two station startup transformers. The applicable emergency procedure incorrectly called for manually tripping the turbine generator output breaker after trip, which did not allow for auto transfer to the startup transformer.
2. The diesel generators started and DG 2 supplied power to the D1 bus.
3. Diesel generator DG 1 tripped on overspeed due to the improper setting of the governor high speed limit switch.
4. Normal off-site power was returned within 11 seconds on the "B" bus and 25 seconds on the "A" bus.

Corrective action:

The applicable emergency procedure was revised to preclude manual tripping of the turbine generator breakers after a turbine trip.

Design purpose of failed system or component:

-

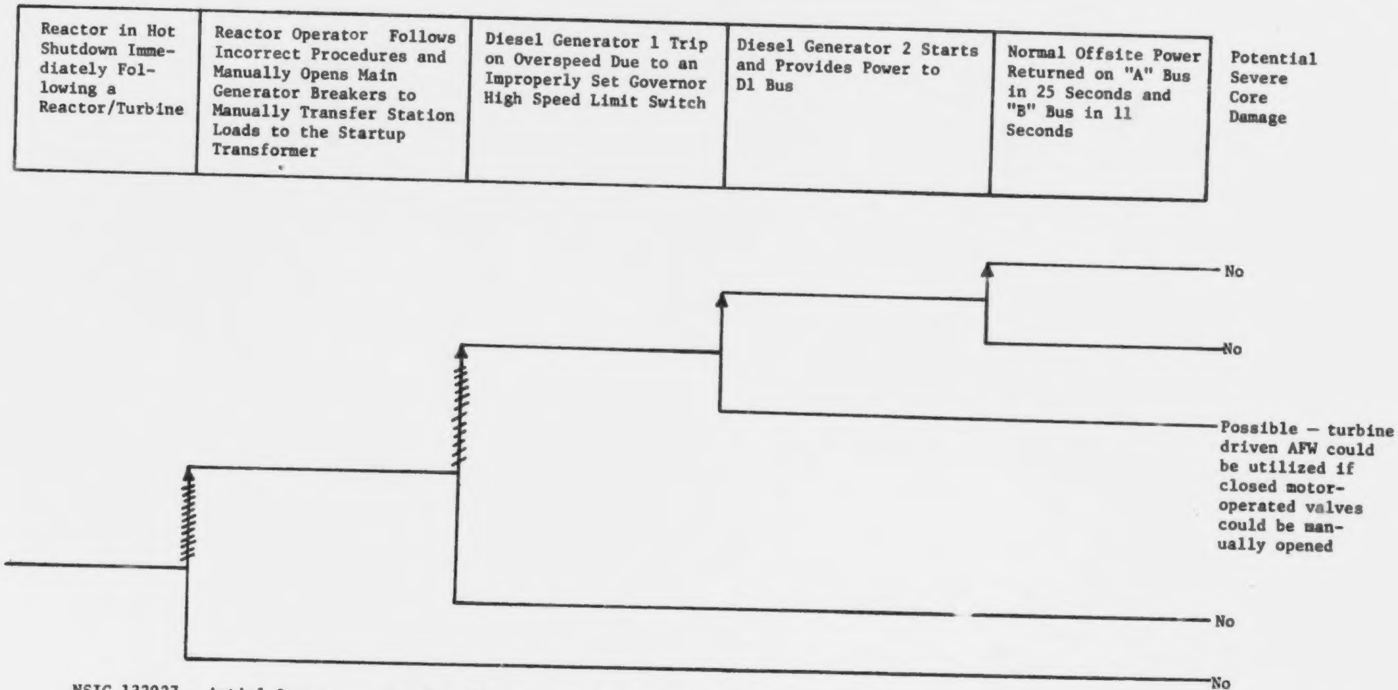
Unavailability of system per WASH 1400:* Loss of offsite power: 2×10^{-5}/hr

Unavailability of component per WASH 1400:* -

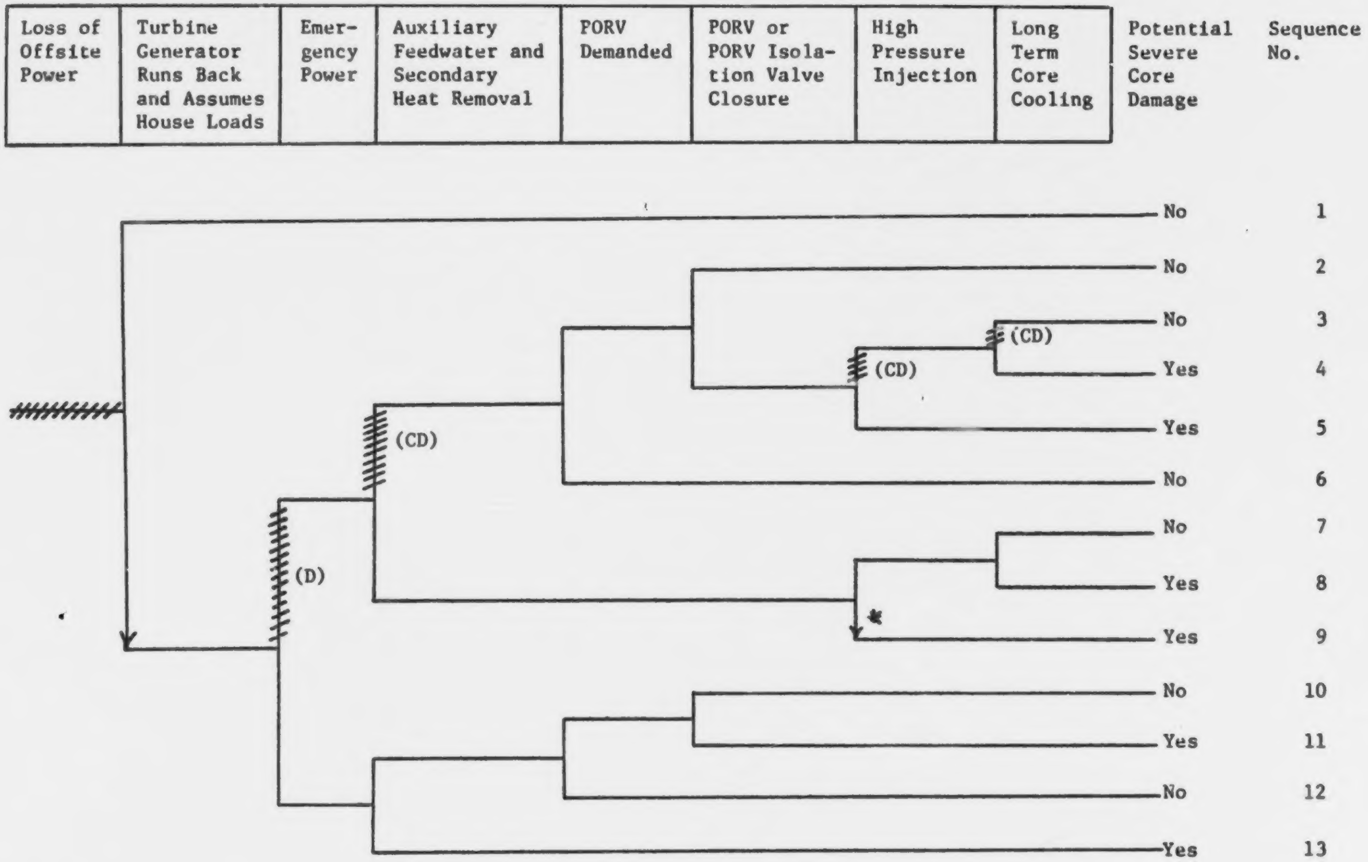
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

BEST DOCUMENT AVAILABLE


Reactor in Hot Shutdown Immediately Following a Reactor/Turbine
Reactor Operator Follows Incorrect Procedures and Manually Opens Main Generator Breakers to Manually Transfer Station Loads to the Startup Transformer
Diesel Generator 1 Trip on Overspeed Due to an Improperly Set Governor High Speed Limit Switch
Diesel Generator 2 Starts and Provides Power to D1 Bus
Normal Offsite Power Returned on "A" Bus in 25 Seconds and "B" Bus in 11 Seconds
Potential Severe Core Damage
No
No
Possible — turbine driven AFW could be utilized if closed motor-operated valves could be manually opened
No
No

NSIC 132927 — Actial Occurrence for Both 13.8 kv Buses De-Energized at Davis-Besse 1


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
(D)
(CD)
(CD)
(CD)
*
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
Yes 9
No 10
Yes 11
No 12
Yes 13

NSIC 132927 — Sequence of Interest for Both 13.8 kv Buses De-Energized at Davis-Besse 1

[*] Not included in operating procedures.

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 132927

DATE OF LER: December 16, 1977

DATE OF EVENT: November 29, 1977

SYSTEM INVOLVED: Electric Power

COMPONENT INVOLVED: Diesel Generator

CAUSE: Operator incorrectly attempted manual transfer to startup transformers, incorrect setting of diesel generator governor high speed limit switch.

SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE: Operator error induced loss of offsite power with subsequent diesel generator overspeed trip.

REACTOR NAME: Davis-Besse 1

DOCKET NUMBER: 50-346

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 906 MWe

REACTOR AGE: .30 yr

VENDOR: Babcock & Wilcox. Co.

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Toledo Edison Co.

LOCATION: 21 miles east of Toledo, Ohio

DURATION: N/A

PLANT OPERATING CONDITION: immediately following trip

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; [✓ (b)] failed to start; [✓ (c)] made inoperable; (d) ____________

DISCOVERY METHOD: operational event

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 132942

Date: December 22, 1977

Title: Both Diesel Generators Inoperable at Prairie Island 1

The failure sequence was:

1. During a refueling outage, an Analog Protection Functional Test was in progress on the "white" instrument bus (one of four 120 VAC safety-related vital buses). During this test, the "red" inverter was incorrectly removed from service for modifications. During the transfer of the "red" vital bus to its alternate source, power was momentarily lost.
2. This resulted in the unblocking of the safety injection logic and safety injection was initiated on low pressurizer pressure and level.
3. The diesel generators started as required.

(See attached page)

Corrective action:

Procedures were revised and additional training provided.

Design purpose of failed system or component:

The diesel-generators provide power to safety-related equipment when offsite power and the unit generator are not available.

Unavailability of system per WASH 1400:* Emergency power: 1×10^{-2}/D

Unavailability of component per WASH 1400:* Diesel generator: 3×10^{-2}/D

---

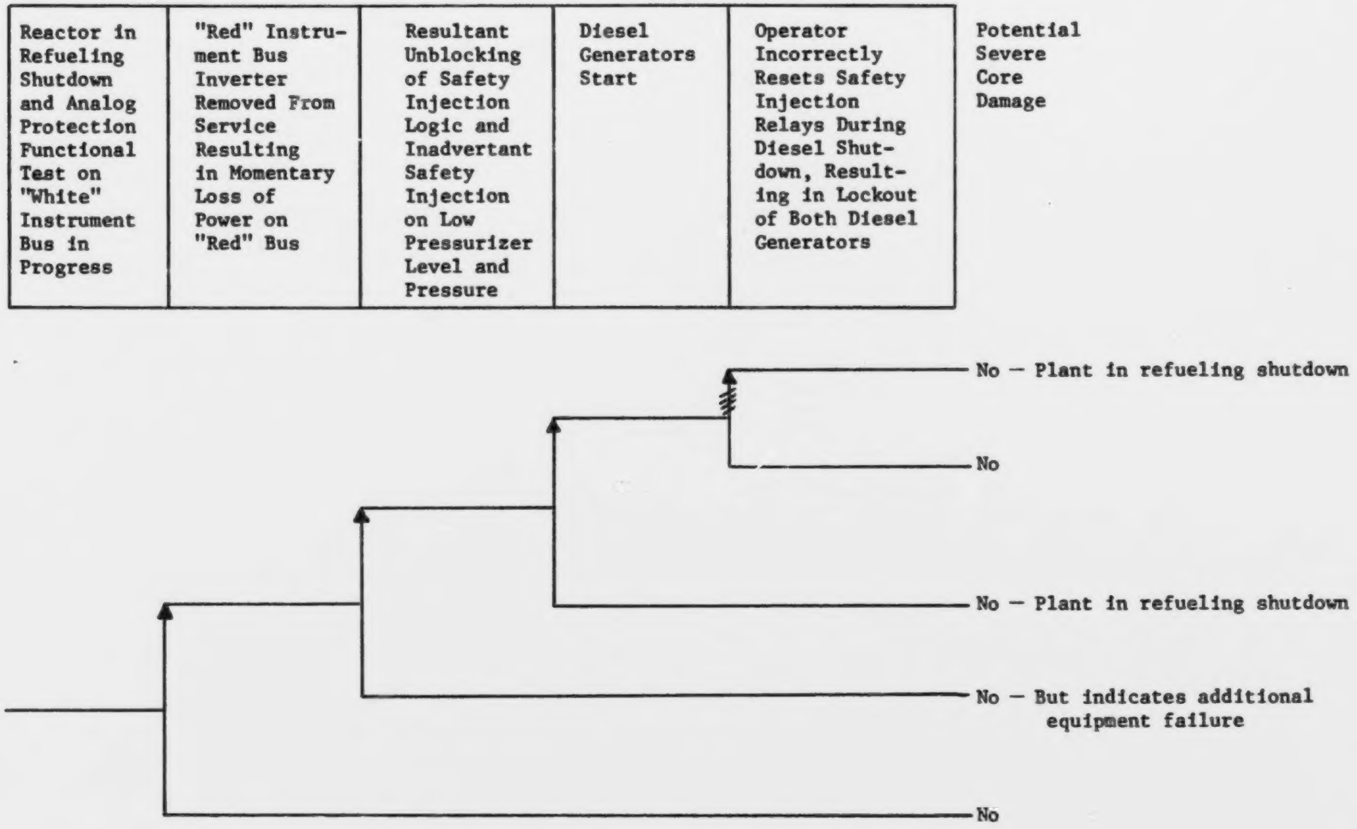
*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

4/19 BEST DOCUMENT AVAILABLE

[redacted]

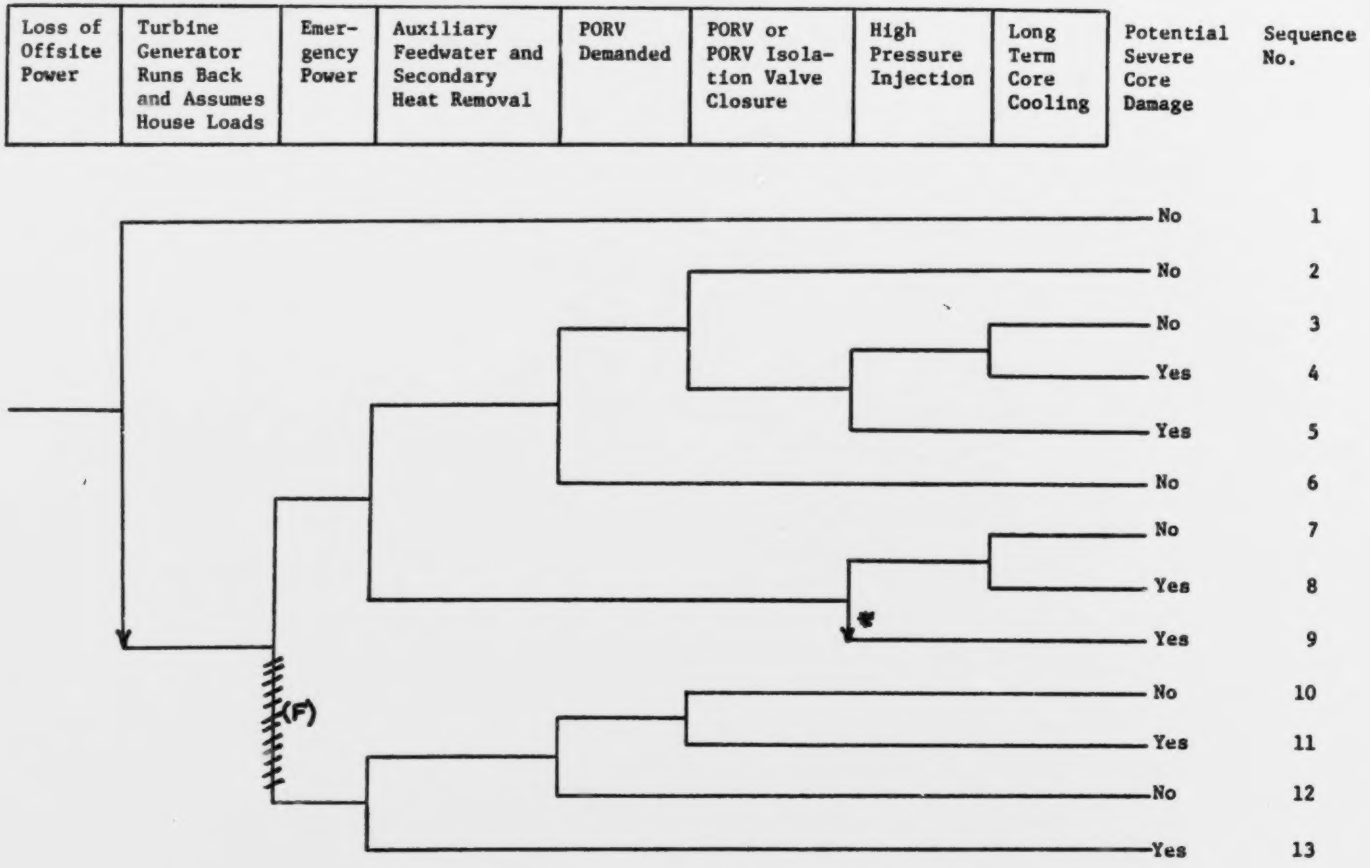
The failure sequence was: (continued)

4. While preparing to secure the diesel generators, the operator failed to reset the safety injection relays prior to stopping the diesels, as required. This resulted in "start failures" for both diesels and the subsequent activation of the shutdown relays.
5. The shutdown relays were reset locally (approximately a ten minute delay) and the diesel generators were returned to service.


Reactor in Refueling Shutdown and Analog Protection Functional Test on "White" Instrument Bus in Progress
"Red" Instrument Bus Inverter Removed From Service Resulting in Momentary Loss of Power on "Red" Bus
Resultant Unblocking of Safety Injection Logic and Inadvertant Safety Injection on Low Pressurizer Level and Pressure
Diesel Generators Start
Operator Incorrectly Resets Safety Injection Relays During Diesel Shutdown, Resulting in Lockout of Both Diesel Generators
Potential Severe Core Damage
No — Plant in refueling shutdown
No
No — Plant in refueling shutdown
No — But indicates additional equipment failure
No

NSIC 132942 — Actual Occurrence for Both Diesel Generators Unavailable at Prairie Island 1

BEST DOCUMENT AVAILABLE


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
(F)
*
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
Yes 9
No 10
Yes 11
No 12
Yes 13

NSIC 132942 — Sequence of Interest for Both Diesel Generators Unavailable at Prairie Island 1

*Not included in operating procedures

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 132942

DATE OF LER: December 22, 1977

DATE OF EVENT: December 9, 1977

SYSTEM INVOLVED: Emergency power

COMPONENT INVOLVED: Diesel generators

CAUSE: Incorrect shutdown procedure performed - human error

SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE: Diesel generator lockout following inadvertant safety injection during refueling

REACTOR NAME: Prairie Island 1

DOCKET NUMBER: 50-282

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 530 MWe

REACTOR AGE: 4.0 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Pioneer

OPERATORS: Northern States Power

LOCATION: 28 miles SE of Minneapolis, Minnesota

DURATION: 17 hours

PLANT OPERATING CONDITION: Refueling shutdown

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________________

DISCOVERY METHOD: Operational event

COMMENT:

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 132943

Date: December 16, 1977

Title: Complete Loss of Offsite Power Occurs at Palisades

The failure sequence was:

1. During normal operation with the reactor at 85% power, switchyard bus "R" became de-energized, causing a complete loss of offsite power and resulting in a loss of main condenser cooling water flow.
2. The reactor was manually tripped.
3. Both diesel generators started and provided power to safety-related loads.

Corrective action;

None; the cause of the "R" bus loss was under investigation.

Design purpose of failed system or component:

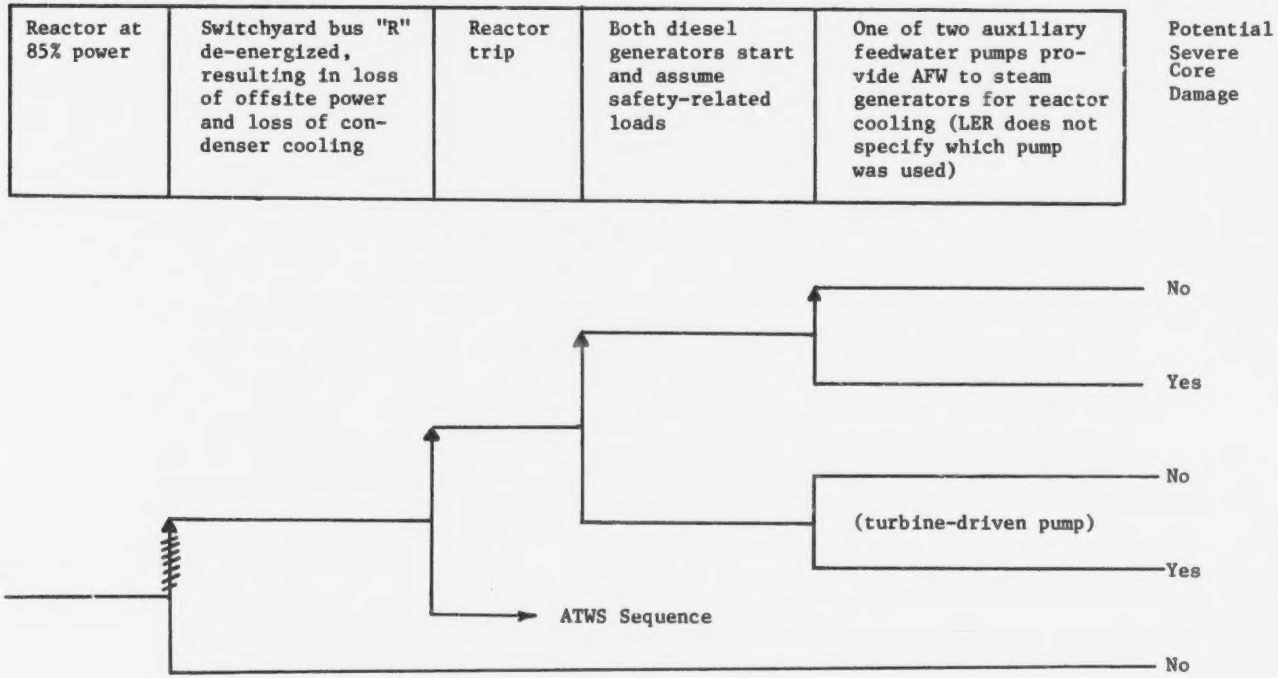
Offsite power provides the preferred source of electric power to plant equipment when the unit generator is not in operation. The condenser circulating water pumps are normally powered from the off-site source.

Unavailability of system per WASH 1400:* loss of offsite power: 2×10^{-5}/hr

Unavailability of component per WASH 1400:* —

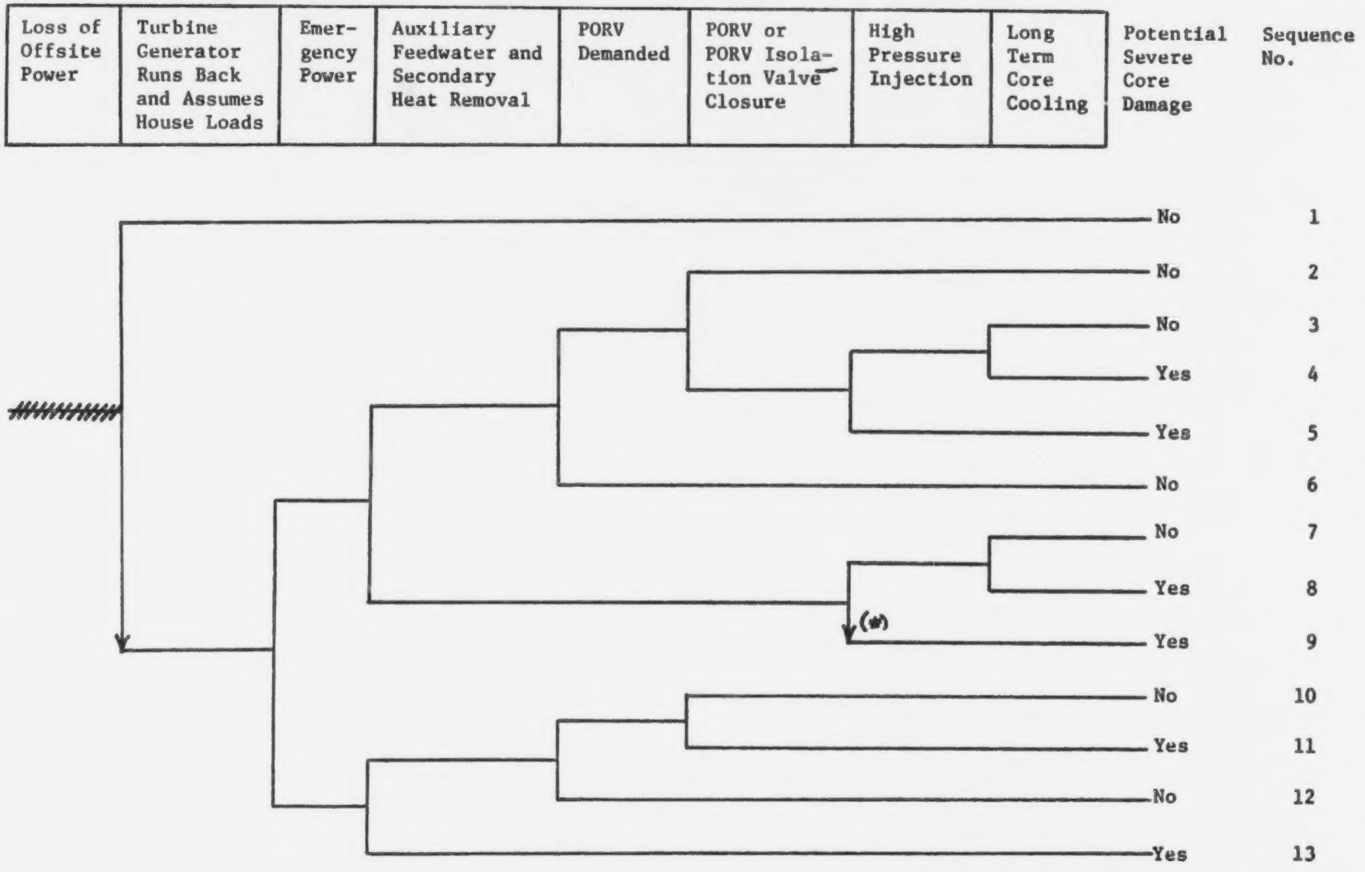
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor at 85% power
Switchyard bus "R" de-energized, resulting in loss of offsite power and loss of condenser cooling
Reactor trip
Both diesel generators start and assume safety-related loads
One of two auxiliary feedwater pumps provide AFW to steam generators for reactor cooling (LER does not specify which pump was used)
Potential Severe Core Damage
No
Yes
No
(turbine-driven pump)
Yes
ATWS Sequence
No

NSIC 132943 — Actual Occurrence for Complete Loss of Offsite Power at Palisades

425


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
(*)
Yes 9
No 10
Yes 11
No 12
Yes 13

NSIC 132943 — Sequence of Interest for Complete Loss of Offsite Power at Palisades

*Not included in mitigation procedures.

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 132943

DATE OF LER: December 16, 1977

DATE OF EVENT: November 25, 1977

SYSTEM INVOLVED: offsite power

COMPONENT INVOLVED: switchyard bus "R"

CAUSE: bus trip from unknown causes

SEQUENCE OF INTEREST: reactor trip with loss of offsite power

ACTUAL OCCURRENCE: reactor trip with loss of offsite power

REACTOR NAME: Palisades

DOCKET NUMBER: 50-255

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 805 MWe

REACTOR AGE: 6.8 yr

VENDOR: Combustion Engineering

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Consumers Power Co.

LOCATION: 5 miles south of South Haven, Mich.

DURATION: N/A

PLANT OPERATING CONDITION: 85% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ________________

DISCOVERY METHOD: during operation

COMMENT: See also 132958

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 132947

Date: December 2, 1977

Title: Diesel Generator Fails to Start at Dresden 2

The failure sequence was:

1. Diesel generator 2/3 was inoperable due to the following reasons: (a) its cooling water pump was out of service and (b) the diaphragm in the air regulator for the air starter was damaged.
2. Diesel generator number 2 failed to start twice for unknown reasons.
3. The following was done prior to the third starting attempt: (a) the fuel oil prime pump was momentarily actuated, (b) the dual fuel oil filter was switched over to the idle filter, and (c) they lay shaft lever that controls the fuel injectors racks was actuated.
4. After these actions the diesel successfully started.

Corrective action:

The cause of failure was not determined. The diesel generator test frequency was increased from once per month to once per week.

Design purpose of failed system or component:

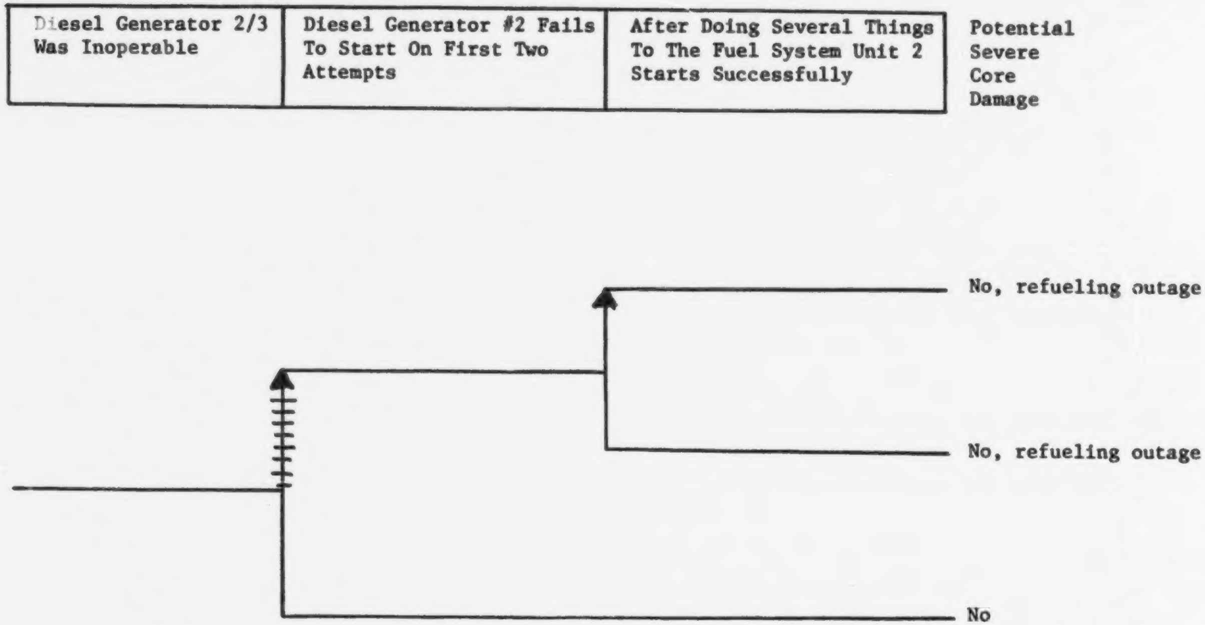
The electric power system provides a.c. power when offsite sources are unavailable.

Unavailability of system per WASH 1400:* Emergency Electric Power: 10^{-3}/D

Unavailability of component per WASH 1400:* Diesel failure to start: 3×10^{-2}/D

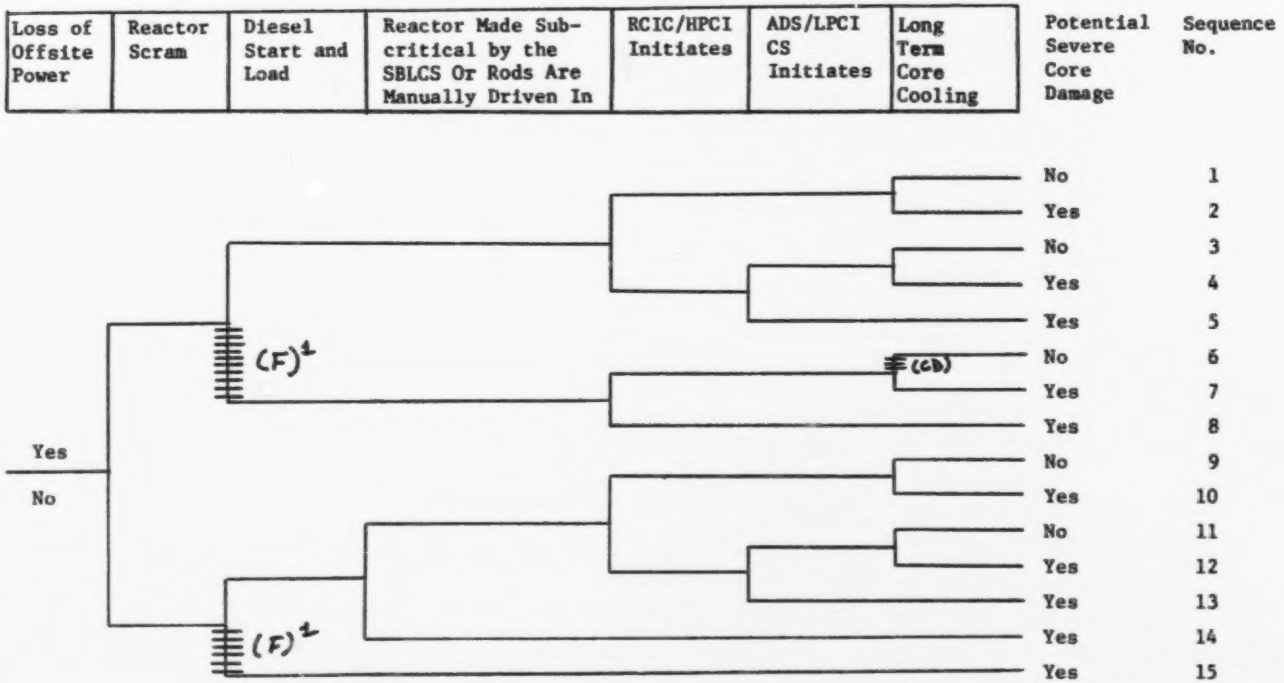
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Diesel Generator 2/3 Was Inoperable
Diesel Generator #2 Fails To Start On First Two Attempts
After Doing Several Things To The Fuel System Unit 2 Starts Successfully
Potential Severe Core Damage
No, refueling outage
No, refueling outage
No

NSIC 132947 — Actual Occurrence of a Diesel Generator Fails to Start at Dresden 2

BEST DOCUMENT AVAILABLE


Loss of Offsite Power
Reactor Scram
Diesel Start and Load
Reactor Made Subcritical by the SBLCS Or Rods Are Manually Driven In
RCIC/HPCI Initiates
ADS/LPCI CS Initiates
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
(F)1
(F)1
(CD)
No 1
Yes 2
No 3
Yes 4
Yes 5
No 6
Yes 7
Yes 8
No 9
Yes 10
No 11
Yes 12
Yes 13
Yes 14
Yes 15

NSIC 132947 — Sequence of Interest of a Diesel Generator Fails to Start at Dresden 2

[1] Success results if, step 3 of "The Failure Sequence Was:" is successful during the LOOP

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 132947

DATE OF LER: December 15, 1977

DATE OF EVENT: December 2, 1977

SYSTEM INVOLVED: Emergency Electric Power

COMPONENT INVOLVED: Diesel Generators

CAUSE: Unknown but not human related

SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE: Diesel Generator Fails to Start at Dresden 2

REACTOR NAME: Dresden 2

DOCKET NUMBER: 50-237

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 794 MWe

REACTOR AGE: 7.9 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: Sargent & Lundy

OPERATORS: Commonwealth Edison

LOCATION: Nine miles from Morris, Ill.

DURATION: 12 (a) hours

PLANT OPERATING CONDITION: refueling outage

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________________

DISCOVERY METHOD: testing

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 132958

Date: December 11, 1977

Title: Complete Loss of Offsite Power Occurs at Palisades

The failure sequence was:

1. With the reactor at 100% power, the "R" switchyard bus de-energized because of a spurious signal from the bus stripping relay, resulting in a complete loss of offsite power and consequent loss of main condenser cooling.
2. The reactor was manually tripped.
3. Both diesel generators started and provided power to safety-related equipment.

Corrective action;

1. The "R" bus tripping scheme was modified to minimize loss of the bus due to spurious action of the "R" bus stripping relay. The specific cause of the stripping relay trip had not been determined.

Design purpose of failed system or component:

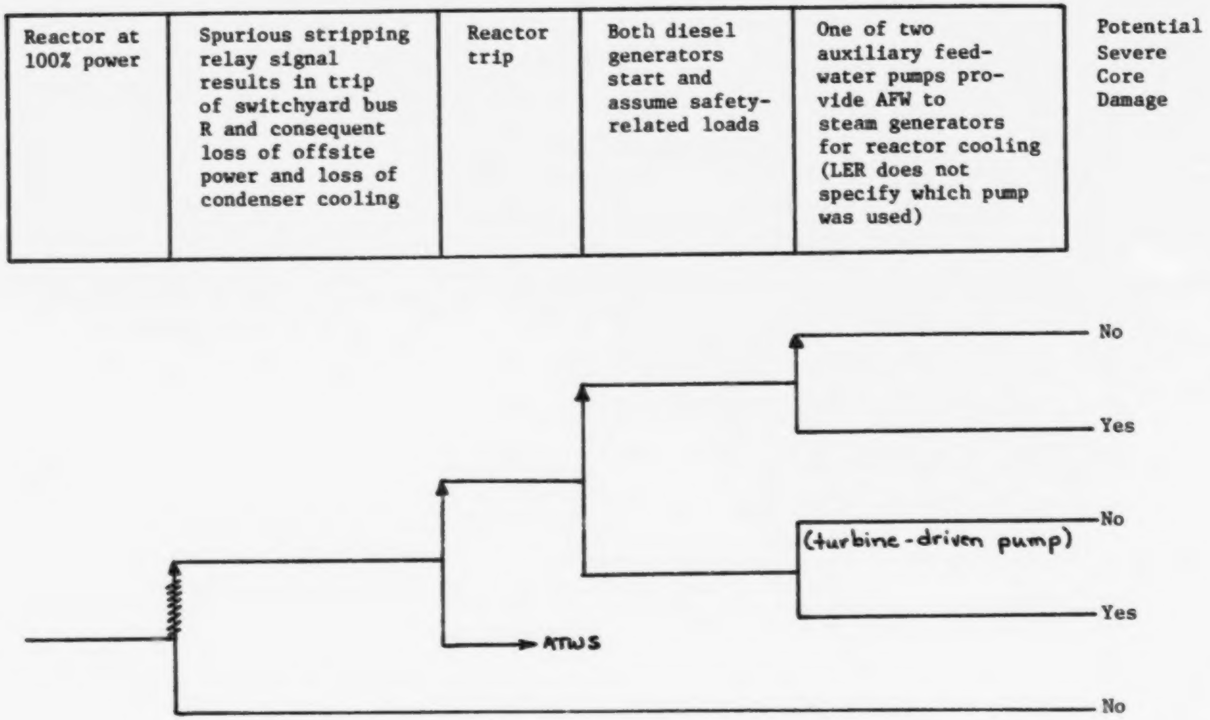
1. Off-site power provides the preferred source of electric power to plant equipment when the unit generator is not in operation. The condenser circulating water pumps are normally powered from the off-site power source.

Unavailability of system per WASH 1400:* loss of offsite power: 2 × 10^{-5}/hr

Unavailability of component per WASH 1400:* –

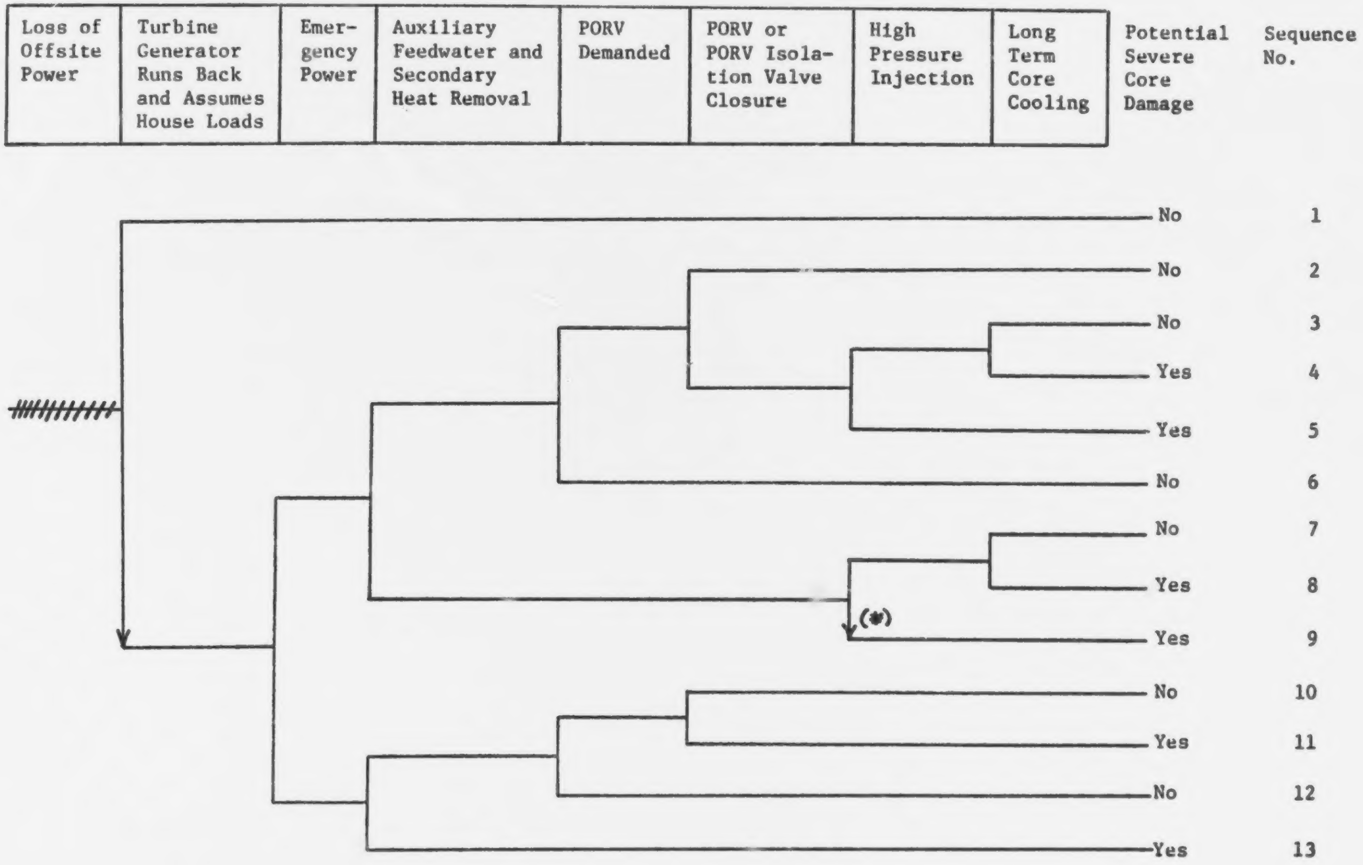
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor at 100% power
Spurious stripping relay signal results in trip of switchyard bus R and consequent loss of offsite power and loss of condenser cooling
Reactor trip
Both diesel generators start and assume safety-related loads
One of two auxiliary feedwater pumps provide AFW to steam generators for reactor cooling (LER does not specify which pump was used)
Potential Severe Core Damage
No
Yes
No
(turbine-driven pump)
Yes
ATWS
No

NSIC 132958 — Actual Occurrence for Complete Loss of Offsite Power at Palisades

BEST DOCUMENT AVAILABLE



Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
(*)
Yes 9
No 10
Yes 11
No 12
Yes 13

NSIC 132958 — Sequence of Interest for Complete Loss of Offsite Power at Palisades
*Not included in mitigation procedures.

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 132958

DATE OF LER: December 21, 1977

DATE OF EVENT: December 11, 1977

SYSTEM INVOLVED: off-site power

COMPONENT INVOLVED: switchyard bus "R"

CAUSE: spurious stripping relay signal

SEQUENCE OF INTEREST: loss of offsite power

ACTUAL OCCURRENCE: reactor trip with loss of offsite power

REACTOR NAME: Palisades

DOCKET NUMBER: 50-255

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 805 MWe

REACTOR AGE: 6.8 yr

VENDOR: Combustion Engineering

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Consumers Power Co.

LOCATION: 5 miles south of South Haven, Mich.

DURATION: N/A

PLANT OPERATING CONDITION: 100% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ________________

DISCOVERY METHOD: during operation

COMMENT: See also 132943

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 132979

Date: August 10, 1977

Title: Both Diesel Generator Fuel Supply Valves Found Shut at Millstone 2

The failure sequence was:

1. During a routine inspection with the plant in cold shutdown, the fuel supply valves on both diesel generators were found shut. The cause of valve closure was not known.
2. Both fuel supply valves were reopened.

Corrective action;

1. The valve manual operators were chain locked in the open position.

Design purpose of failed system or component:

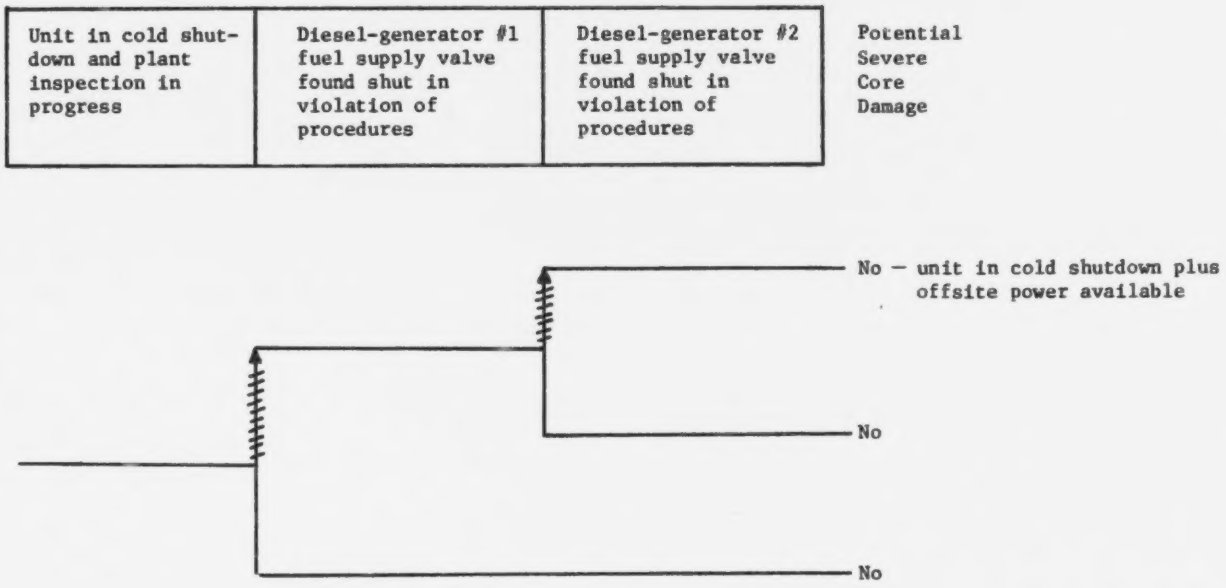
The diesel generators provide on-site emergency power for safety-related loads in the event offsite power and the unit generator are unavailable.

Unavailability of system per WASH 1400:* emergency power: 3×10^{-2}/D

Unavailability of component per WASH 1400:* diesel generators: 10^{-2}/D

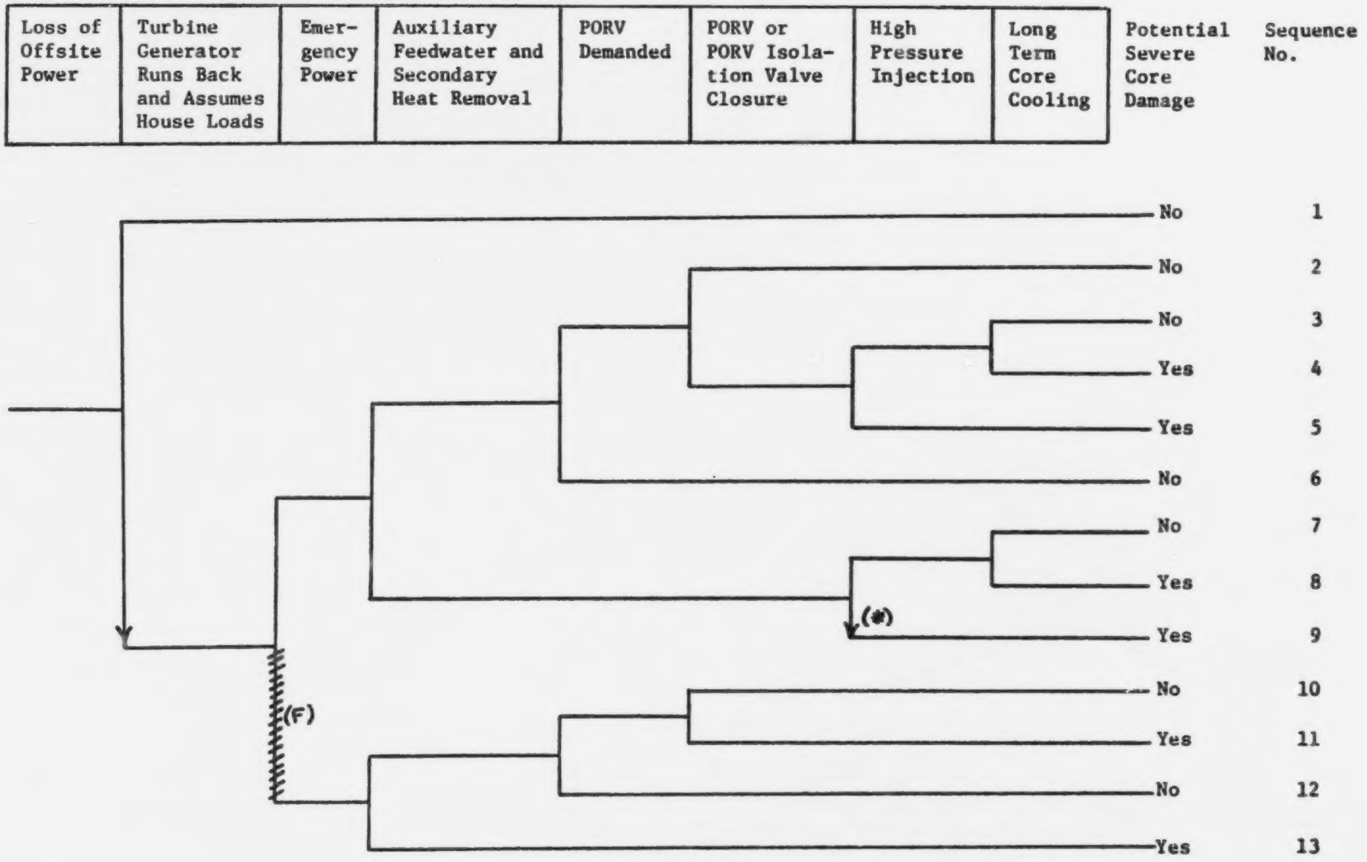
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Unit in cold shutdown and plant inspection in progress
Diesel-generator #1 fuel supply valve found shut in violation of procedures
Diesel-generator #2 fuel supply valve found shut in violation of procedures
Potential Severe Core Damage
No — unit in cold shutdown plus offsite power available
No
No

NSIC 132979 — Actual Occurrence for Both Diesel Generator Fuel Supply Valves Found Shut at Millstone 2

4/37


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
(*)
Yes 9
(F)
No 10
Yes 11
No 12
Yes 13

NSIC 132979 — Sequence of Interest for Both Diesel-Generator Fuel Supply Valves Found Shut at Millstone 2
*Not included in mitigation procedures.

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 132979

DATE OF LER: August 10, 1977 (Original report date June 14, 1977)

DATE OF EVENT: May 15, 1977

SYSTEM INVOLVED: on-site emergency power

COMPONENT INVOLVED: diesel-generators

CAUSE: fuel supply valves shut, human error

SEQUENCE OF INTEREST: loss of offsite power

ACTUAL OCCURRENCE: fuel supply valves discovered to be shut during an inspection

REACTOR NAME: Millstone 2

DOCKET NUMBER: 50-336

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 870 MWe

REACTOR AGE: 2.0 yr

VENDOR: Combustion Engineering

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Northeast Nuclear Energy Co.

LOCATION: 5 miles SW of New London, Conn.

DURATION: 180(a) hours

PLANT OPERATING CONDITION: cold shutdown

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ________________

DISCOVERY METHOD: routine inspection

COMMENT: _

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: **133004**

Date: **June 13, 1977**

Title: **Three Out of Four Diesels Inoperable at Peach Bottom Atomic Power Station**

The failure sequence was:

1. **Unit 2 was shutdown for maintenance and unit 3 was operating at 100% power.**
2. **Diesel generator E1 was taken out of service for annual maintenance.**
3. **About 3 hours later the operator noticed a diesel trouble light on both diesels E3 and E4 and dispatched an operator to investigate the problem.**
4. **After performing other tasks (no priority was given) the operator discovered that both air receivers on both E3 and E4 were empty and their associated compressors were tripped on thermal overload.**
5. **The thermal overload switches were reset and the air receivers were charged.**
6. **Later in the day another operator was dispatched to check the status of the compressors and the air receivers. The compressors were again found tripped on thermal overload; however, the air receivers were found to have sufficient air.**

(see attached sheet)

Corrective action:

1. **The sectionalizing valve was locked closed.**
2. **The leaking check valves were replaced.**

Design purpose of failed system or component:

1. **The diesel generators provide emergency power should all offsite sources be lost.**
2. **The air receivers and compressors provide compressed air to start the diesel engine.**

Unavailability of system per WASH 1400:* **- Emergency Power: $1\cdot10^{-3}$/D**

Unavailability of component per WASH 1400:* **diesel generators: 3×10^{-2}/D**

---

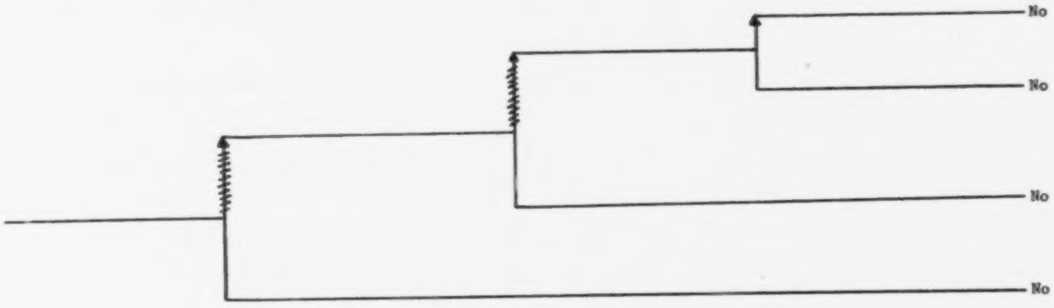
[*] Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

[redacted]

The failure sequence was: (continued)

7. The problem was traced to two causes: (1) A sectionalizing valve which allows the air starting system of the diesels to be interconnected, was left partially opened. (2) The air receiver check valves were found to be leaking. The compressors tripped on thermal overload as they tried to maintain air pressure in the air receiver tanks.

Unit 2 down for maintenance. Unit 3 at 100% power, diesel generator E1 down for maintenance. Trouble light on both diesel generators E3 & E4	Operator finds air starting tanks empty due to leaky check valves and charging compressors tripped on thermal overload. He resets compressors which begin to recharge cylinders	A follow-up inspection reveals that the compressors are again tripped. Sectionalizing valve found partially opened. Valve closed & compressors reset.	Leaky check valves replaced & sectionalizing valve locked closed	Potential Severe Core Damage



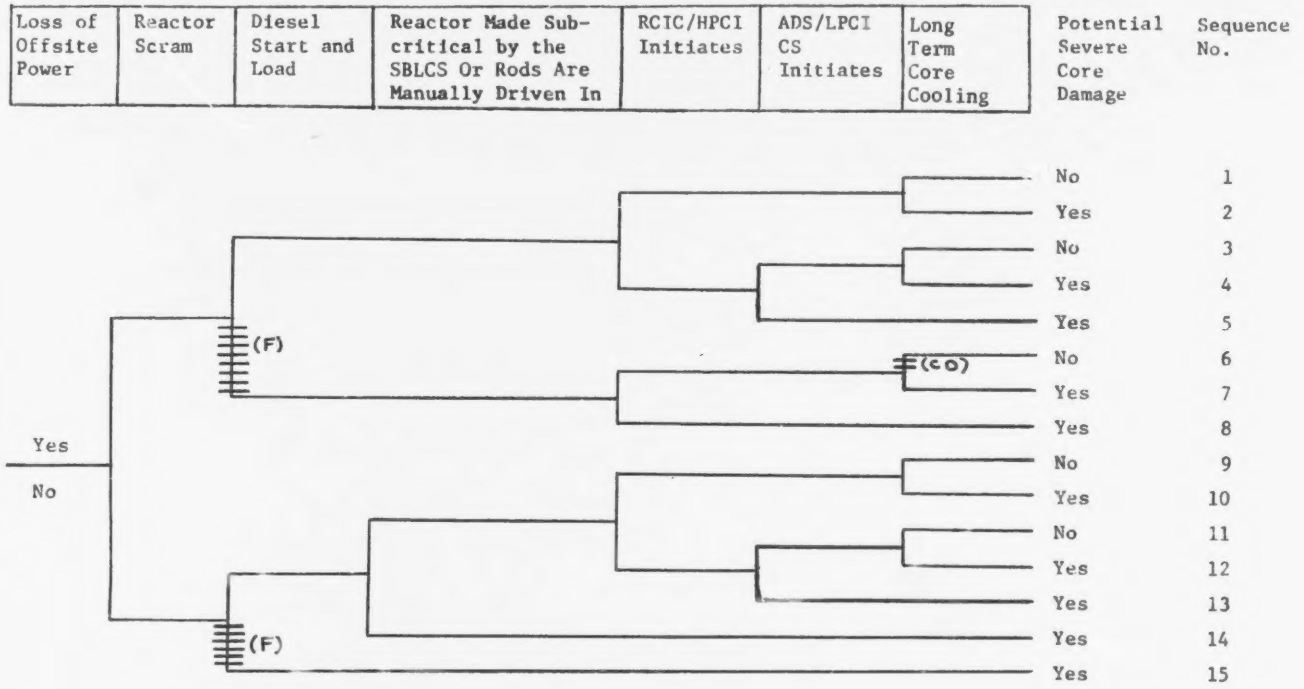
No

No - plant not experiencing a transient, however the emergency power system will be severely degraded on a continuing basis.

No - plant not experiencing a transient, but emergency power is severely degraded

No - plant not experiencing a transient, but emergency power is severely degraded

NSIC 133004 – Actual Occurrence of Three Out of Four Diesels Inoperable at Peach Bottom Atomic Power Station


Loss of Offsite Power
Reactor Scram
Diesel Start and Load
Reactor Made Sub-critical by the SBLCS Or Rods Are Manually Driven In
RCIC/HPCI Initiates
ADS/LPCI CS Initiates
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
(F)
(CO)
(F)
No 1
Yes 2
No 3
Yes 4
Yes 5
No 6
Yes 7
Yes 8
No 9
Yes 10
No 11
Yes 12
Yes 13
Yes 14
Yes 15

NSIC 133004 - Sequence of Interest for Three Out of Four Diesels Inoperable at Peach Bottom Atomic Power Station

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 133004

DATE OF LER: June 30, 1977

DATE OF EVENT: June 13, 1977

SYSTEM INVOLVED: electric power

COMPONENT INVOLVED: diesel generators

CAUSE: common mode failure of the diesels, (human error)

SEQUENCE OF INTEREST: loss of offsite power

ACTUAL OCCURRENCE: three out of four diesels inoperable at Peach Bottom

REACTOR NAME: Peach Bottom 3

DOCKET NUMBER: 50-278

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 1065 MWe

REACTOR AGE: 2.8 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Philadelphia Electric Company

LOCATION: 19 miles S of Lancaster

DURATION: 24 (a) hours

PLANT OPERATING CONDITION: 100% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________

DISCOVERY METHOD: alarm

COMMENT:

BEST DOCUMENT AVAILABLE 444

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 133706

Date: December 11, 1977

Title: Control of Both Auxiliary Feed Pump Turbines Lost at Davis-Besse 1

The failure sequence was:

1. During testing, control over both auxiliary feedwater pump turbines was lost, rendering the pumps inoperable.
2. Control of the turbine for pump 1-1 was lost due to mechanical binding in the governor.
3. Control of the turbine for pump 1-2 was lost due to blown control power supply fuses for the speed changer motor.

Corrective action;

1. On pump 1-1, the roll pin which was binding was repositioned, cleaned, and lubricated.
2. On pump 1-2, the blown fuses were replaced, as were two relays suspected of sticking and causing the blown fuses.

Design purpose of failed system or component:

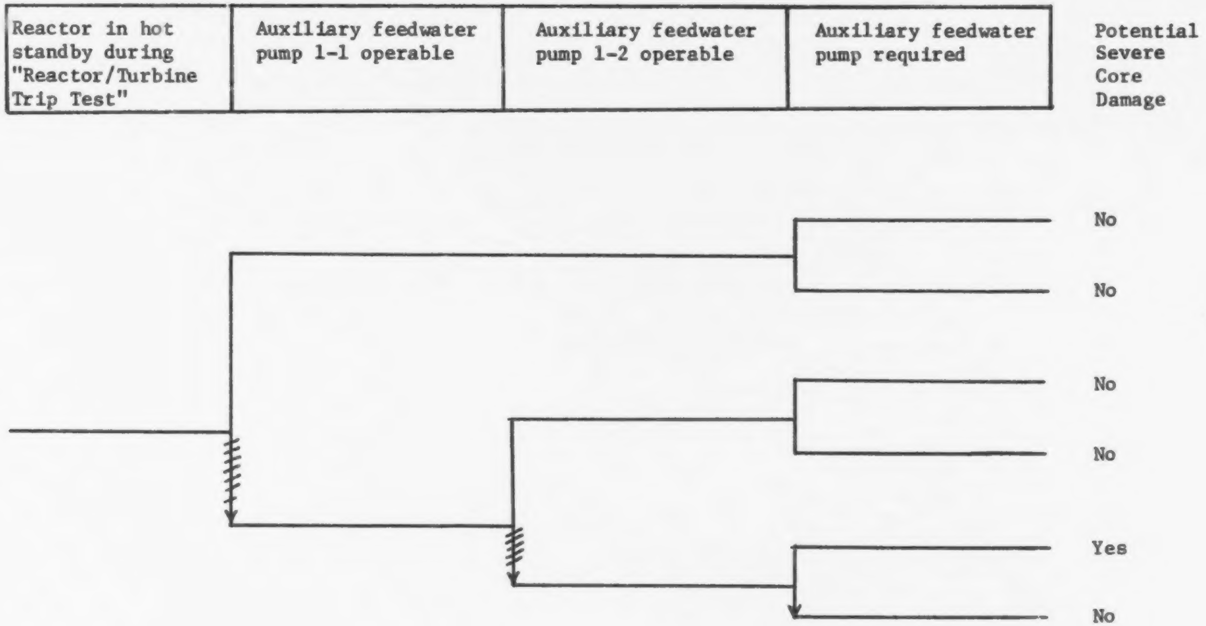
1. The auxiliary feedwater pumps provide RCS heat removal when the main feed system is unavailable.

Unavailability of system per WASH 1400:* Auxiliary feedwater: $3.3 \times 10^{-5}/D$

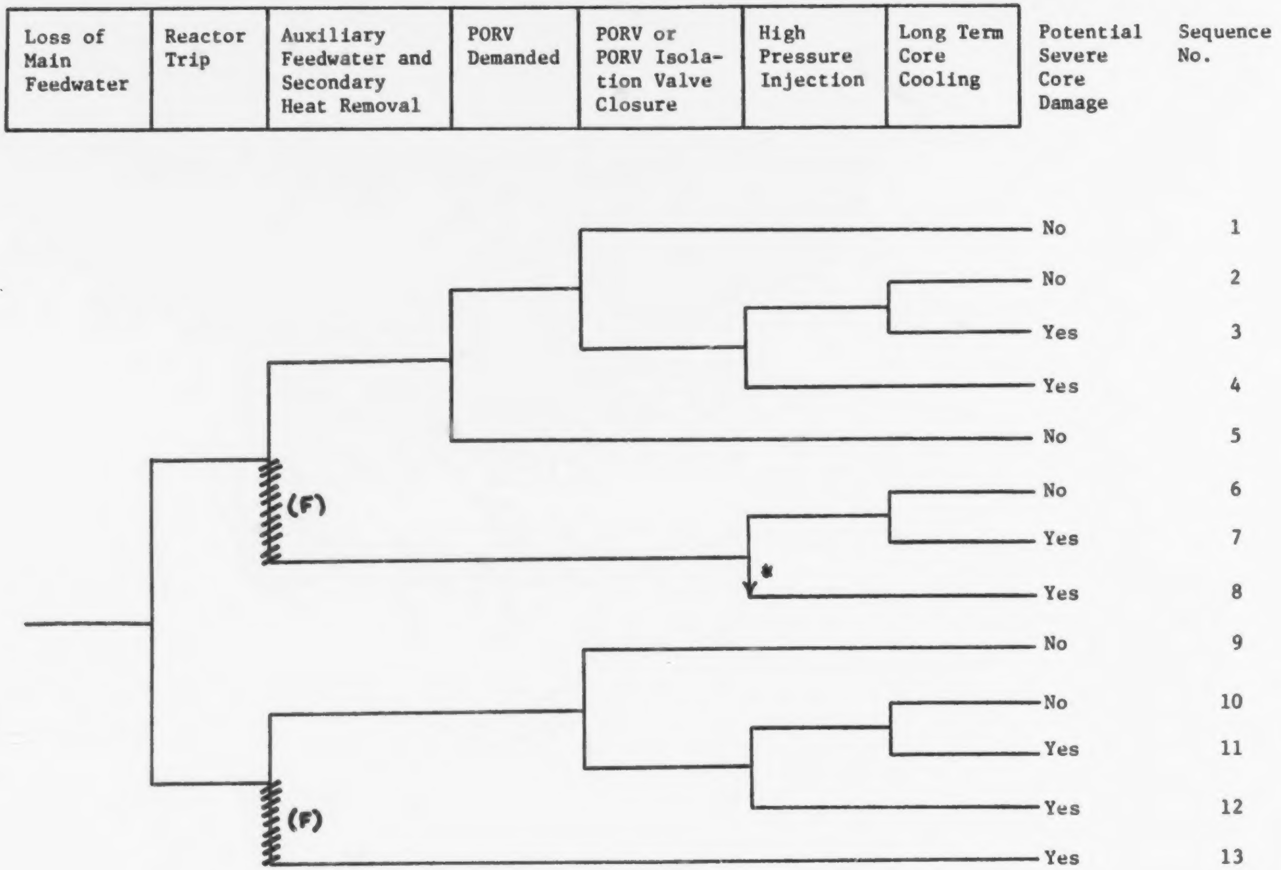
Unavailability of component per WASH 1400:* Pumps, failure to start $1 \times 10^{-3}/D$

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor in hot standby during "Reactor/Turbine Trip Test"
Auxiliary feedwater pump 1-1 operable
Auxiliary feedwater pump 1-2 operable
Auxiliary feedwater pump required
Potential Severe Core Damage
No
No
No
No
Yes
No

NSIC 133706 — Actual Occurrence for Control of Both Auxiliary Feed Pump Turbines Lost at Davis-Besse 1


Loss of Main Feedwater
Reactor Trip
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
Yes 3
Yes 4
No 5
No 6
Yes 7
Yes 8
No 9
No 10
Yes 11
Yes 12
Yes 13
(F)
(F)
*

NSIC 133706 — Sequence of Interest for Control of Both Auxiliary Feed Pump Turbines Lost at Davis-Besse 1

*not included in mitigation procedures

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 133706

DATE OF LER: January 3, 1978

DATE OF EVENT: December 11, 1977

SYSTEM INVOLVED: Auxiliary feedwater

COMPONENT INVOLVED: Auxiliary feedwater pump turbines

CAUSE: Mechanical binding, blown fuses

SEQUENCE OF INTEREST: Loss of feedwater

ACTUAL OCCURRENCE: Auxiliary feedwater pumps inoperable during testing.

REACTOR NAME: Davis-Besse 1

DOCKET NUMBER: 50-346

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 906 MWe

REACTOR AGE: 0.6 yr

VENDOR: Babcock & Wilcox

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Toledo Edison

LOCATION: 21 miles east of Toledo, Ohio

DURATION: 360(a) hours

PLANT OPERATING CONDITION: hot standby

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ________________

DISCOVERY METHOD: During testing

COMMENT: –

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 135006

Date: September 29, 1977

Title: Loss of Offsite Power Occurs at Farley 1

The failure sequence was:

1. The plant was at 4% power and the main generator was being brought to synchronous speed. One of the redundant connections between the two switchyard 230 kV buses was inoperable because of breaker maintenance. A second connection between the buses was open because the main generator circuit breakers were open.
2. Lightning struck the facility in the area of the turbine building and low voltage switchyard during a severe electrical storm. An induced voltage spike actuated a type AR auxiliary relay which opened two circuit breakers, de-energizing the #1 230 kV bus and startup transformer 1B. The relay actuated because of an incomplete ground tie between the substation and the plant and because of incorrect grounding of a shielded cable carrying the AR relay trip circuits. Startup transformer 1A was de-energized by the loss of the #1 bus. (See next page)

Corrective action:

1. Offsite power was restored in approximately one hour.
2. Digital Rod Position Indication power was re-routed from a vital bus.
3. The improper cable ground and cable connections were corrected.

Design purpose of failed system or component:

Offsite power provides the preferred source of power to safety-related components when the main generator is not providing power.

Unavailability of system per WASH 1400:* Loss of offsite power: 2×10^{-5}/hr

Unavailability of component per WASH 1400:* -

---

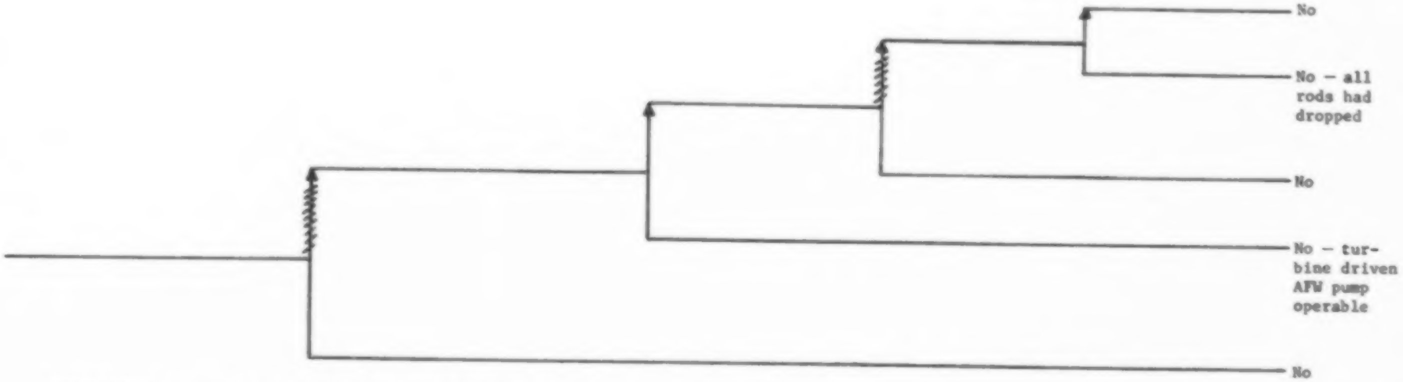
*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

[redacted]

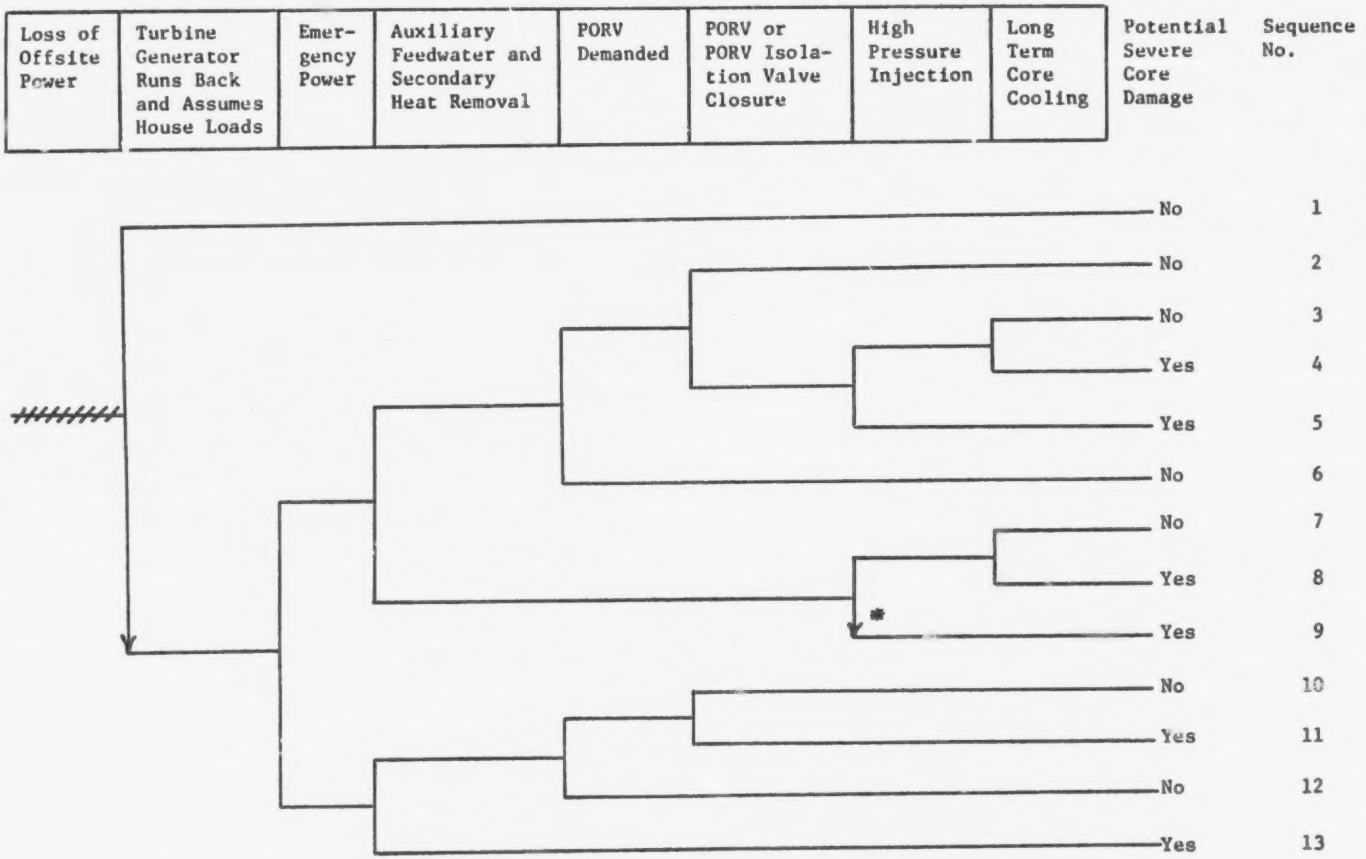
The failure sequence was: (continued)

3. This resulted in the loss of all 4160 volt buses.
4. All unit 1 emergency diesel generators started and assumed safety-related loads.
5. Digital rod position indication was lost since it was powered from a non-vital bus.
6. Because of the lack of digital rod position indication, the operator could not confirm that all rods had dropped, and he initiated emergency boration.
7. Reactor shutdown was verified by observing a negative startup rate indication.

Reactor at 4% Power and Generator Being Brought Up to Speed. One Connection Between the Two 230 kv Buses Because of Breaker Maintenance and the Fact the Main Generator Was Not on Line	Lightning Strike Results in an Auxiliary Relay Actuation Due to Incomplete and Incorrect Circuit Grounding. This Opens Two Circuit Breakers which Causes the De-energization of Both Startup Transformers and the #1 230 kv Bus and Results in a Loss of Power on all 4160 Volt Buses	Both Diesel Generators Start and Provide Power to Safety-Related Loads	Loss of Power on Non-Vital Buses Results in Loss of Digital Rod Position	Operator Initiates Emergency Boration Due to Inability to Determine that All Rods had Dropped	Potential Severe Core Damage


No
No – all rods had dropped
No
No – turbine driven AFW pump operable
No

NSIC 135006 – Actual Occurrence for Loss of Offsite Power Occurs at Farley 1


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
*
Yes 9
No 10
Yes 11
No 12
Yes 13

NSIC 135006 — Sequence of Interest for Loss of Offsite Power Occurs at Farley 1

*Not included in mitigation procedures.

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 135006

DATE OF LER: September 29, 1977

DATE OF EVENT: September 16, 1977

SYSTEM INVOLVED: Offsite Power

COMPONENT INVOLVED: Circuit Breakers

CAUSE: Inadvertent opening due to actuation of an auxiliary relay because of lightning induced voltage spikes.

SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE: Loss of offsite power

REACTOR NAME: Farley 1

DOCKET NUMBER: 50-348

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 829 MWe

REACTOR AGE: .10 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Bechtel/SSI

OPERATORS: Alabama Power Company

LOCATION: 24 mi SE of Dothan, Alabama

DURATION: N/A

PLANT OPERATING CONDITION: 4% power

SAFETY FEATURE TYPE OF FAILURE: [✓ (a)] inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ________________

DISCOVERY METHOD: operational event

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 135077

Date: November 8, 1977

Title: BIT Thermo Well Coupling Ruptures at Robinson 2

The failure sequence was:

1. During an inadvertant safety injection, the operators received a Boron injection tank low level and isolated the tank,
2. A thermo well coupling on the tank had failed, resulting in a one inch hole in the tank.

Corrective action;

1. The entire tank was replaced.

Design purpose of failed system or component:

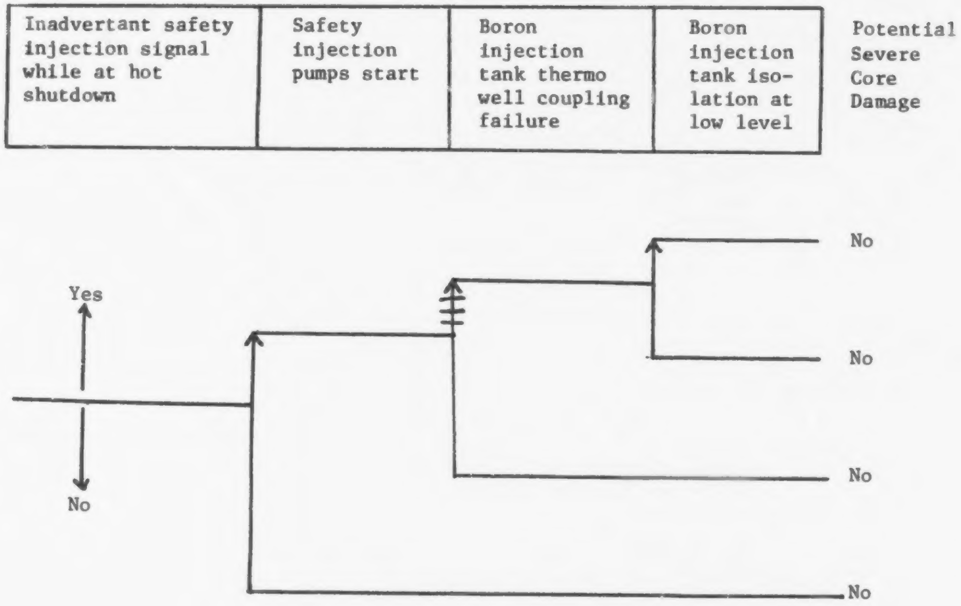
1. The tank stores 12% boric acid for injection with the RCS during safety injection. This boric acid returns the core to a subcritical condition following a steam line break.

Unavailability of system per WASH 1400:* High Pressure Injection System: 8.6×10^{-3}

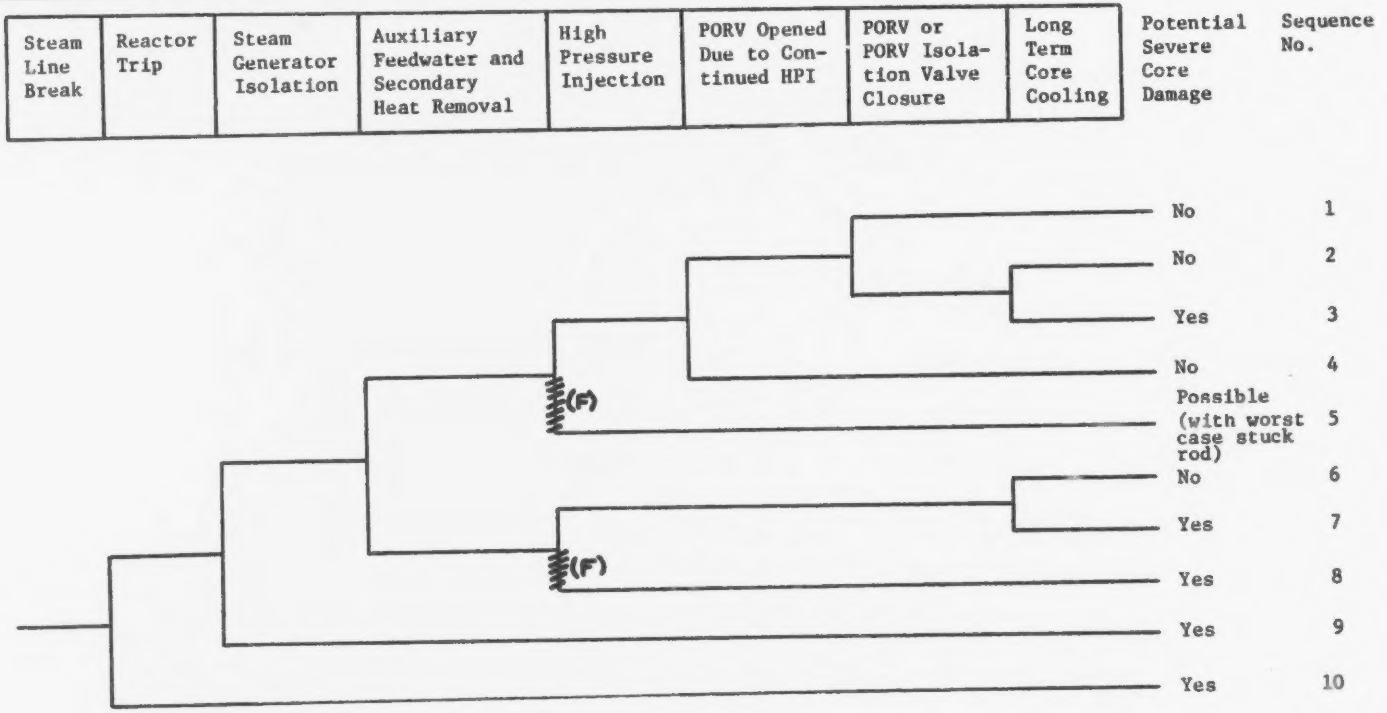
Unavailability of component per WASH 1400:* BIT Rupture: ϵ

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Inadvertant safety injection signal while at hot shutdown
Safety injection pumps start
Boron injection tank thermo well coupling failure
Boron injection tank isolation at low level
Potential Severe Core Damage
Yes
No
No
No
No
No

NSIC 135077 — Actual Occurrence for Boron Injection Tank Thermo Well Coupling Ruptures at Robinson 2


Steam Line Break
Reactor Trip
Steam Generator Isolation
Auxiliary Feedwater and Secondary Heat Removal
High Pressure Injection
PORV Opened Due to Continued HPI
PORV or PORV Isolation Valve Closure
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
Yes 3
No 4
Possible (with worst case stuck rod) 5
No 6
Yes 7
Yes 8
Yes 9
Yes 10
(F)
(F)

NSIC 135077 – Sequence of Interest for Boron Injection Tank Thermo Well Coupling Rupture at Robinson 2

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 135077

DATE OF LER: November 8, 1977

DATE OF EVENT: October 26, 1977

SYSTEM INVOLVED: Safety injection

COMPONENT INVOLVED: Boron Injection Tank

CAUSE: Failure of a thermo well coupling on the BIT tank

SEQUENCE OF INTEREST: Steam Line Break

ACTUAL OCCURRENCE: Thermo well coupling failure following inadvertent safety injection

REACTOR NAME: Robinson 2

DOCKET NUMBER: 50-261

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 700 MWe

REACTOR AGE: 7.2 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Ebasco

OPERATORS: Carolina Power & Light Co.

LOCATION: **5 miles NW of Hartsville,** SC

DURATION: 360(a) hours

PLANT OPERATING CONDITION: Hot standby

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ________________

DISCOVERY METHOD: during inadvertant safety injection

COMMENT: The couplings were sensitized and an improper setup of the coupling was used during fabrication.

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 135079

Date: December 8, 1977

Title: All Safety Injection Pumps Inoperable During Heatup at Robinson 2

The failure sequence was:

1. While the reactor was shut down, the SI pumps were tested and determined to be **operable.** The pump breakers were then racked out as required for overpressure protection during cold shutdown.
2. During heatup (∿4 hrs later) the pump breakers were not racked in, as required before going above 200°F due to a misunderstanding of the procedures.
3. This fact was discovered while completing a minimum equipment list and the breakers were racked in (five hours after they were required to be racked in).

Corrective action;

1. The breakers were racked in.
2. Operating procedures were revised as necessary to require the pumps to be in service before exceeding 200°F.

Design purpose of failed system or component:

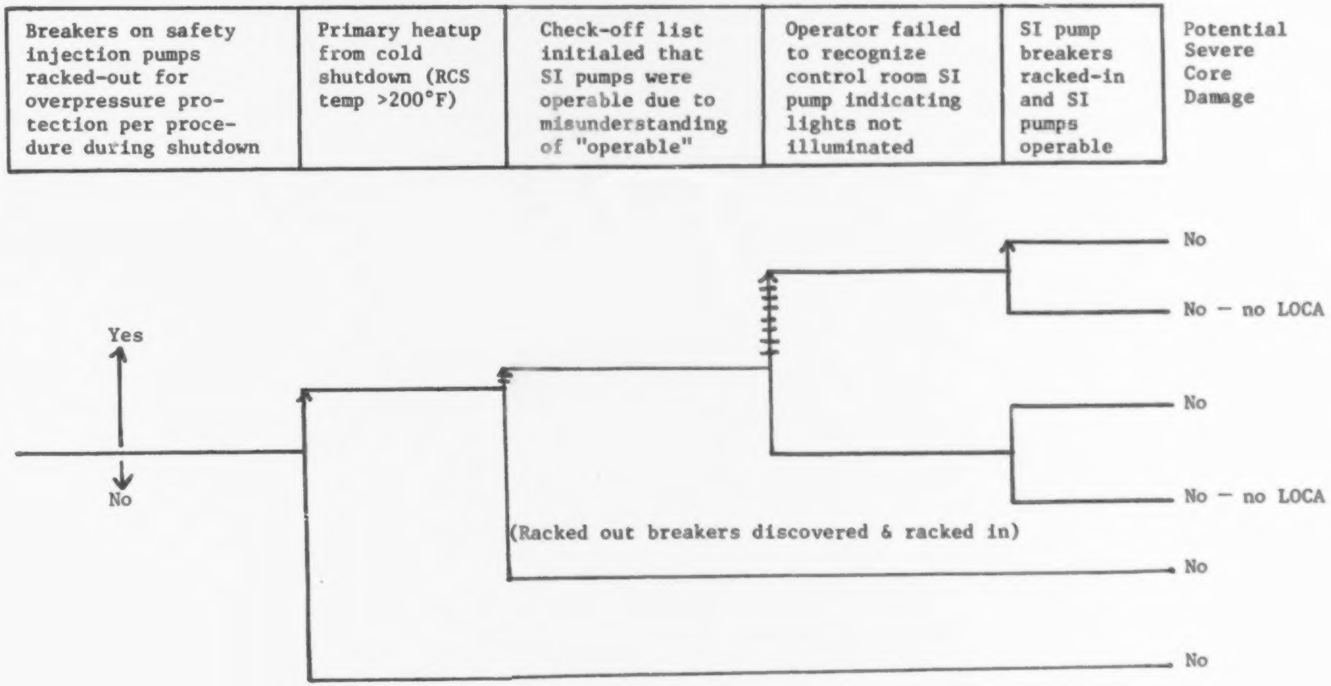
1. The safety injection pumps provide high pressure injection of borated water in the event of a small LOCA.

Unavailability of system per WASH 1400:* High Pressure Injection System: 1.2×10^{-2}/D

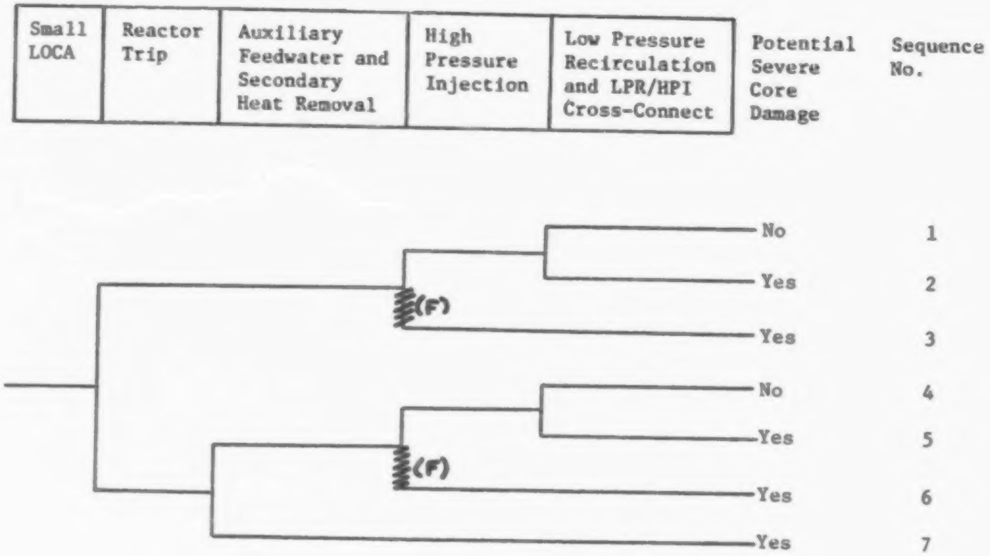
Unavailability of component per WASH 1400:* human error of omission: 3×10^{-3}/D

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Breakers on safety injection pumps racked-out for overpressure protection per procedure during shutdown
Primary heatup from cold shutdown (RCS temp >200°F)
Check-off list initialed that SI pumps were operable due to misunderstanding of "operable"
Operator failed to recognize control room SI pump indicating lights not illuminated
SI pump breakers racked-in and SI pumps operable
Potential Severe Core Damage
Yes
No
No
No — no LOCA
No
No — no LOCA
(Racked out breakers discovered & racked in)
No
No

NSIC 135079 — Actual Occurrence for Inoperable Safety Injection Pumps at Robinson 2


Small LOCA
Reactor Trip
Auxiliary Feedwater and Secondary Heat Removal
High Pressure Injection
Low Pressure Recirculation and LPR/HPI Cross-Connect
Potential Severe Core Damage
Sequence No.
(F)
(F)
No
1
Yes
2
Yes
3
No
4
Yes
5
Yes
6
Yes
7

NSIC 135079 — Sequence of Interest for Inoperable Safety Injection Pumps at Robinson 2

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 135079

DATE OF LER: December 8, 1977

DATE OF EVENT: November 23, 1977

SYSTEM INVOLVED: Safety injection

COMPONENT INVOLVED: Pumps, breakers

CAUSE: Breakers incorrectly racked out, human error

SEQUENCE OF INTEREST Small LOCA

ACTUAL OCCURRENCE: Misunderstanding of procedures during a heatup, the pump breakers were not racked in prior to exceeding 200°F.

REACTOR NAME: Robinson 2

DOCKET NUMBER: 50-261

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 700 MWe

REACTOR AGE: 7.3 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Ebasco

OPERATORS: Carolina Power & Light Co.

LOCATION: 5 miles NW of Hartsville, SC

DURATION: 5 hours

PLANT OPERATING CONDITION: Hot shutdown

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start;
[✓ (c)] made inoperable; (d) ________________

DISCOVERY METHOD: While completing a minimum equipment list

COMMENT: The fact that the pumps were not in service was indicated by a lack of pump status lights on the control board. The operators did not notice that the lights were out.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: **135932**

Date: **February 6, 1978**

Title: **All Offsite Transmission Lines Lost at Pilgrim 1**

The failure sequence was:

The reactor was operating at 24% power when high winds and ice caused all the transmission lines to fail. This event caused the reactor to trip.

Corrective action:

The insulators were washed with fresh water and the reactor was returned to service. A plant design change was issued to resolve the flashover problem on standoff insulators.

Design purpose of failed system or component:

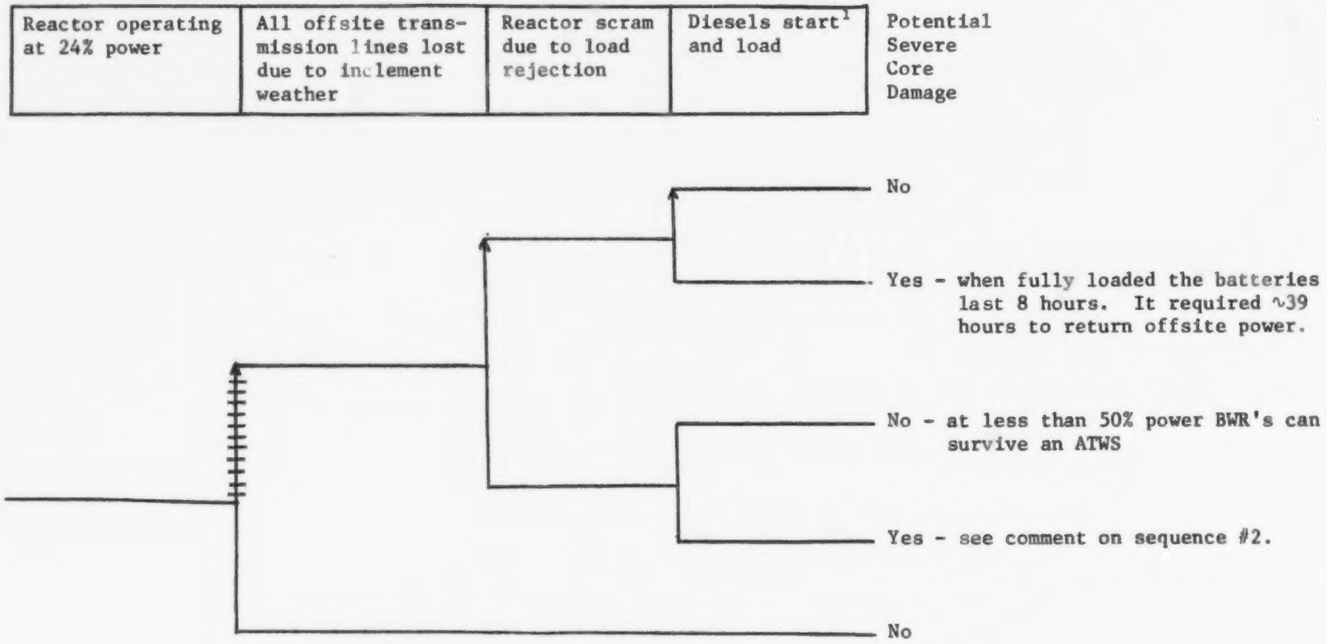
The transmission lines connect Pilgrim 1 with the New England Grid. These lines provide power to the reactor when the station isn't generating.

Unavailability of system per WASH 1400:* **loss of offsite power 2×10^{-5}/hr**

Unavailability of component per WASH 1400:* -

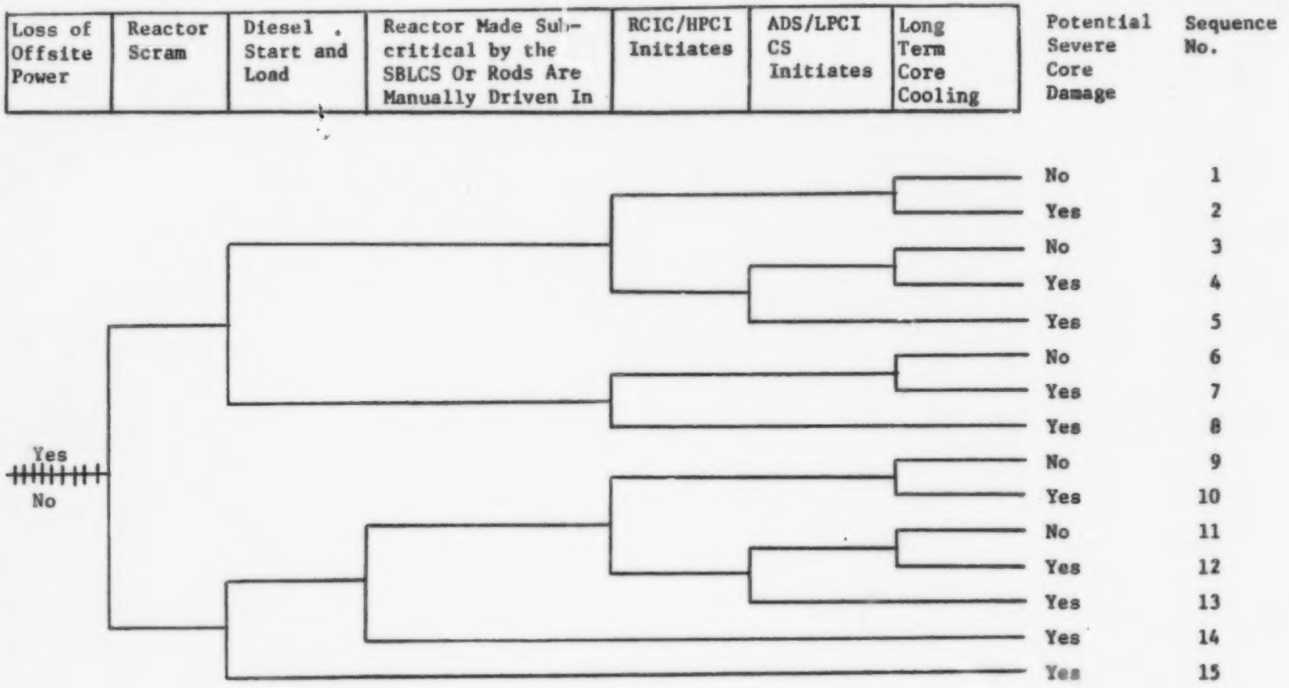
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor operating at 24% power
All offsite transmission lines lost due to inclement weather
Reactor scram due to load rejection
Diesels start[1] and load
Potential Severe Core Damage
No
Yes - when fully loaded the batteries last 8 hours. It required ~39 hours to return offsite power.
No - at less than 50% power BWR's can survive an ATWS
Yes - see comment on sequence #2.
No

NSIC 135932 — Actual Occurrence of the Loss of All Transmission Lines at Pilgrim 1

[1] It is presumed the diesels started and loaded properly.


Loss of Offsite Power
Reactor Scram
Diesel Start and Load
Reactor Made Subcritical by the SBLCS Or Rods Are Manually Driven In
RCIC/HPCI Initiates
ADS/LPCI CS Initiates
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
No 1
Yes 2
No 3
Yes 4
Yes 5
No 6
Yes 7
Yes 8
No 9
Yes 10
No 11
Yes 12
Yes 13
Yes 14
Yes 15

NSIC 135932 — Sequence of Interest of the Loss of All Transmission Lines at Pilgrim 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 135932

DATE OF LER: February 23, 1978

DATE OF EVENT: February 6, 1978

SYSTEM INVOLVED: electric power

COMPONENT INVOLVED: offsite transmission lines

CAUSE: insulator flashover due to inclement weather

SEQUENCE OF INTEREST: loss of offsite power

ACTUAL OCCURRENCE: loss of offsite power

REACTOR NAME: Pilgrim 1

DOCKET NUMBER: 50-293

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 655 MWe

REACTOR AGE: 5.7 yr

VENDOR: GE

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Boston Edison

LOCATION: Four miles SE of Plymouth, Mass.

DURATION: N/A

PLANT OPERATING CONDITION: 25% full power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c) made inoperable]; (d) ____________________

DISCOVERY METHOD: operational event

COMMENT: -

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 137305

Date: April 7, 1978

Title: Auxiliary Feedwater System Failure at Farley 1

The failure sequence was:

1. A low-low water level condition occurred in one steam generator and resulted in a reactor trip.
2. Both motor-driven auxiliary feedwater pumps started.
3. A low-low water level condition occurred in the second steam generator.
4. The turbine driven auxiliary feedwater pump failed to start.
5. The recirculation bypass valves were found open on all auxiliary feedwater pumps, rendering the operating motor-driven pumps ineffective.

Corrective action:

Isolation valves were closed restoring system to operable status.

Design purpose of failed system or component:

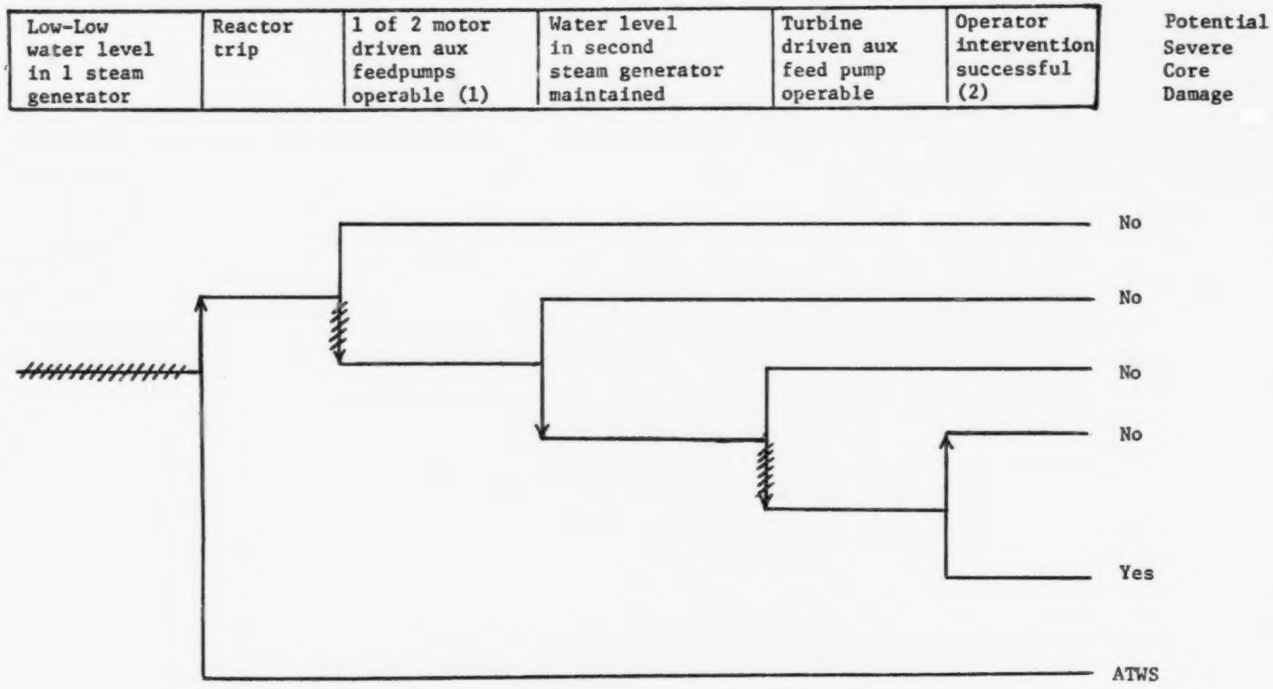
Prevent core damage upon loss of normal feedwater.

Unavailability of system per WASH 1400:* $3.5 \times 10^{-5}/D$

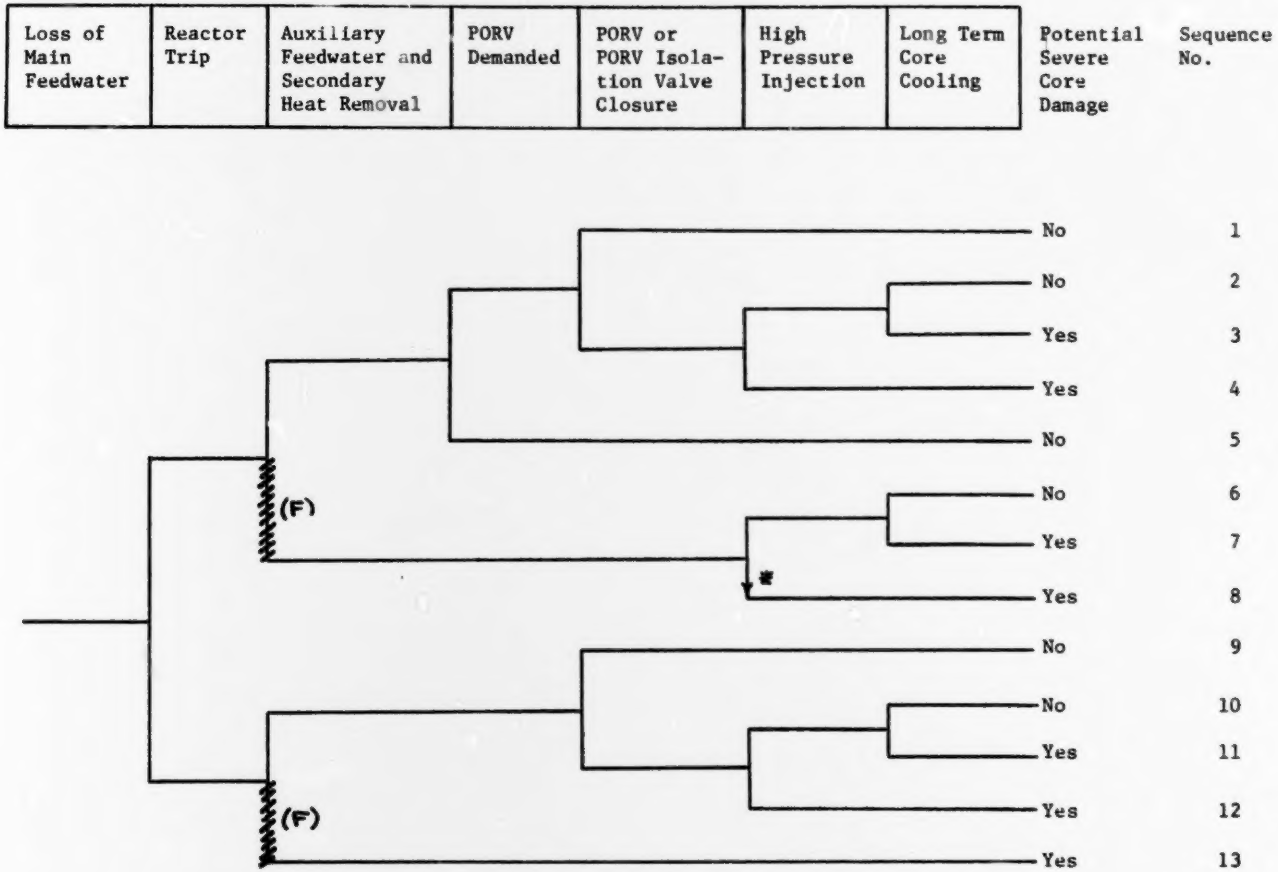
Unavailability of component per WASH 1400:* -

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Low-Low water level in 1 steam generator
Reactor trip
1 of 2 motor driven aux feedpumps operable (1)
Water level in second steam generator maintained
Turbine driven aux feed pump operable
Operator intervention successful (2)
Potential Severe Core Damage
No
No
No
No
Yes
ATWS

NSIC 137305 — Actual Occurrence for Auxiliary Feedwater System Failures (1 & 2). (1) Recirc bypass valves were later found open (resulting in flow reduction) (2) opened a closed turbine trip/throttle valve.


Loss of Main Feedwater
Reactor Trip
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
Yes 3
Yes 4
No 5
No 6
Yes 7
Yes 8
No 9
No 10
Yes 11
Yes 12
Yes 13
(F)
(F)
*

NSIC 137305 — Sequence of Interest for Auxiliary Feedwater System Failures at Farley 1

* not included in mitigation sequence

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 137305

DATE OF LER: April 7, 1978

DATE OF EVENT: March 25, 1978

SYSTEM INVOLVED: Auxiliary feedwater

COMPONENT INVOLVED: Valves

CAUSE: Human error — valves left open

SEQUENCE OF INTEREST: Loss of normal feedwater

ACTUAL OCCURRENCE: Low-low water level in one steam generator

REACTOR NAME: Farley 1

DOCKET NUMBER: 50-348

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 829 MWe

REACTOR AGE: 0.6 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Alabama Power Co.

LOCATION: **24 miles SE of Dothan, Ala.**

DURATION: 360(a) hours

PLANT OPERATING CONDITION: 0% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ________________

DISCOVERY METHOD: Operational event

COMMENT: —

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 137335

Date: March 21, 1978

Title: Operator Error Renders the Isolation Condenser Inoperable with HPCI out for repairs.

The failure sequence was:

1. Tests were being performed on unit three. During these tests, power to the bus feeding the Isolation Condenser was inadvertently switched off.
2. HPCI was unavailable due to maintenance.

Corrective action:

Power was restored to the bus.

Design purpose of failed system or component:

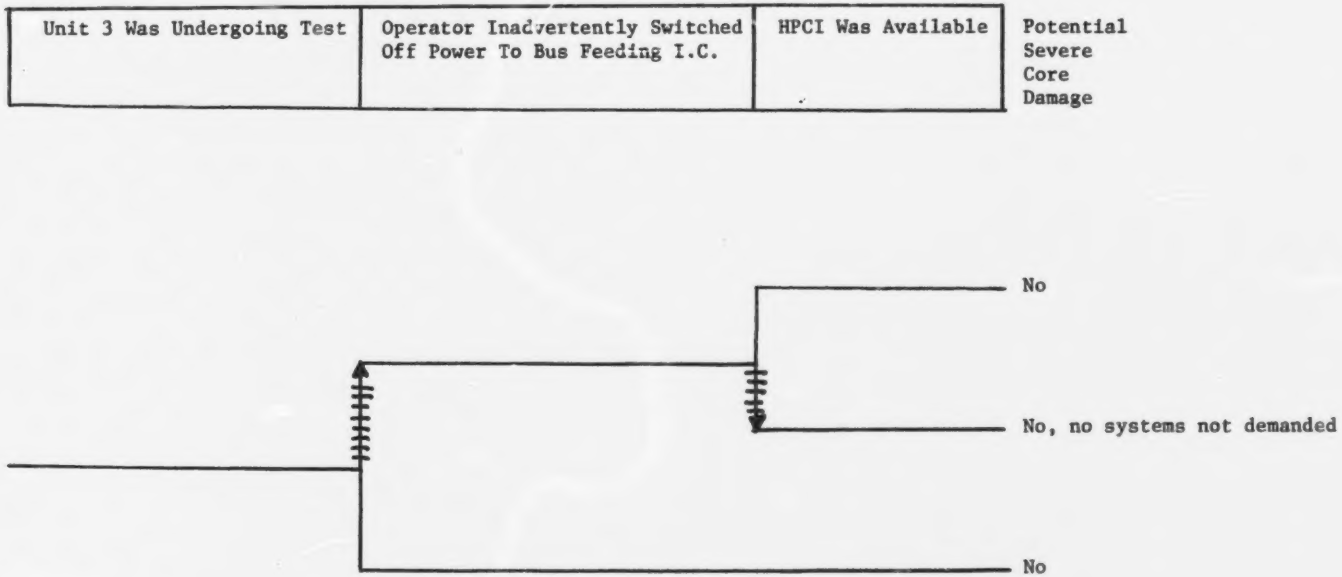
1. The isolation condenser allows for core cooling when the reactor is isolated.
2. HPCI provides for core cooling during a small loca.

Unavailability of system per WASH 1400:* - I.C.: Not covered in Wash-1900
HPCI: $8.8.10^{-2}$/D

Unavailability of component per-WASH 1400:*

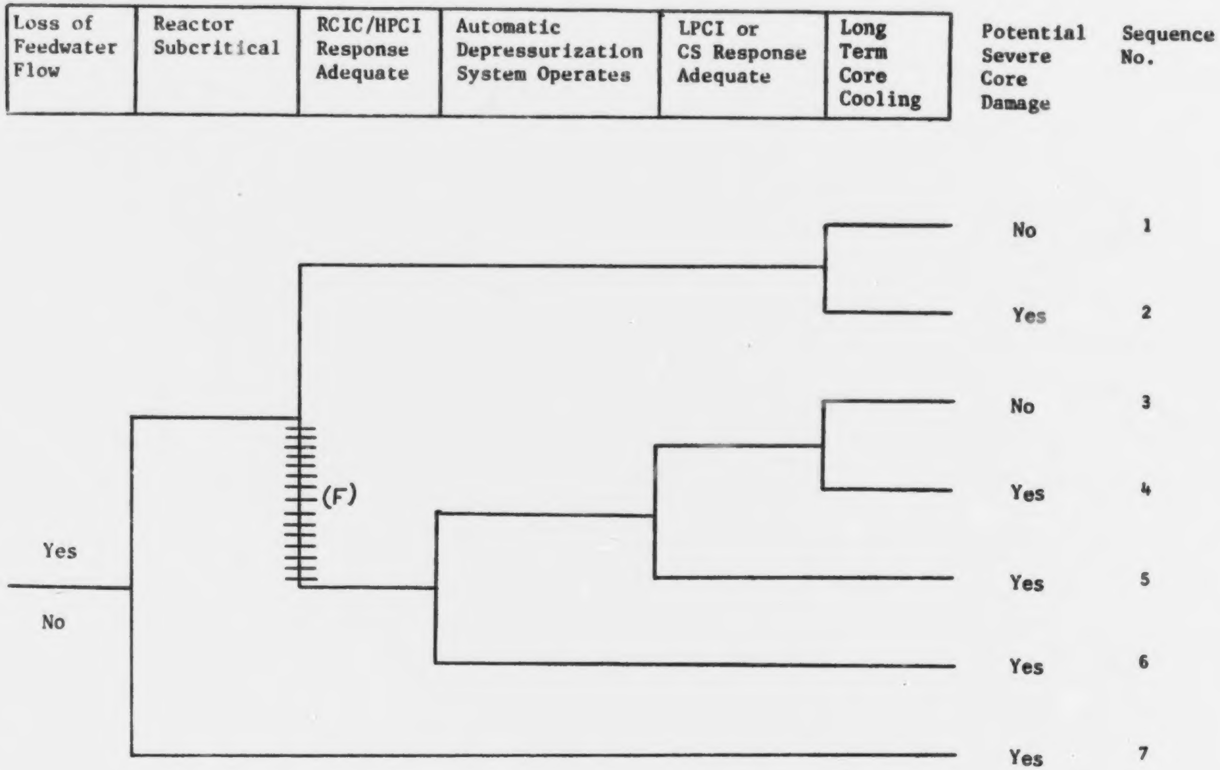
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Unit 3 Was Undergoing Test
Operator Inadvertently Switched Off Power To Bus Feeding I.C.
HPCI Was Available
Potential Severe Core Damage
No
No, no systems not demanded
No

NSIC 137335 — Actual Occurrence of Operator Error Renders the Isolation Condenser Inoperable With HPCI Out for Repairs at Dresden 2

BEST DOCUMENT AVAILABLE


Loss of Feedwater Flow
Reactor Subcritical
RCIC/HPCI Response Adequate
Automatic Depressurization System Operates
LPCI or CS Response Adequate
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
(F)
No 1
Yes 2
No 3
Yes 4
Yes 5
Yes 6
Yes 7

NSIC 137335 — Sequence of Interest for Operator Error Renders IC Inoperable With HPCI Out for Repairs at Dresden 2

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 137335

DATE OF LER: April 4, 1978

DATE OF EVENT: March 21, 1978

SYSTEM INVOLVED: Isolation Condenser, HPCI

COMPONENT INVOLVED: Isolation Condenser Bus.

CAUSE: Operator Error

SEQUENCE OF INTEREST: Loss of Feedwater

ACTUAL OCCURRENCE: Operator Error Renders the Isolation Condenser Inoperable with HPCI Out for Repairs.

REACTOR NAME: Dresden 2

DOCKET NUMBER: 237

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 794 MWe

REACTOR AGE: 8.3 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: Sargent and Lundy

OPERATORS: Commonwealth Edison

LOCATION: Nine miles E of Morris, Ill.

DURATION: 0.75 hours

PLANT OPERATING CONDITION: at power.

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c) made inoperable]; (d) ______________________

DISCOVERY METHOD: Operational Event

COMMENT: Only the abstract was available for review.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 137384

Date: March 18, 1978

Title: Two Makeup Pumps Trip During Testing at Three Mile Island 1

The failure sequence was:

1. During performance of the Emergency Sequence and Power Transfer Test, both makeup pumps failed to start after power was restored to their respective buses. In one case, a lube oil pressure timer relay failed to close, and in the other case, a relay failed to disengage when the pumps initially tripped due to loss of power during the test.

Corrective action;

1. The timer relays were to be replaced during the refueling outage which was in existence during the test.

Design purpose of failed system or component:

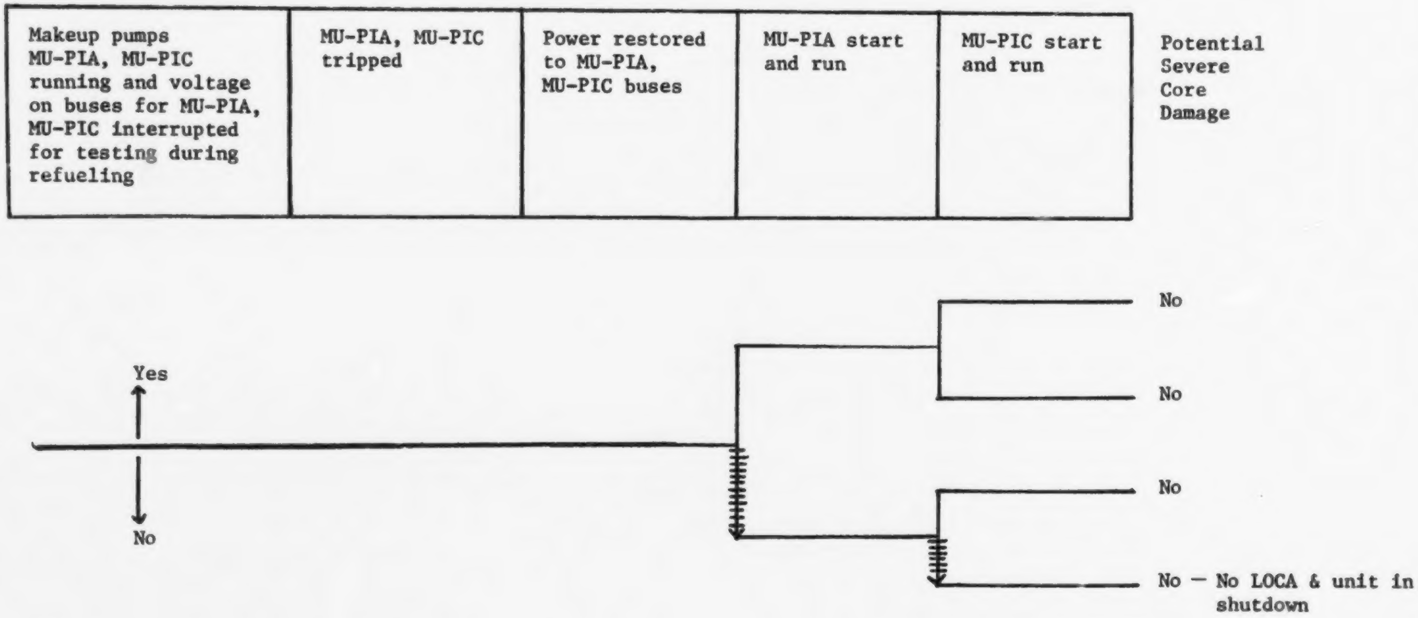
1. The timer relay allows the pump to run for six seconds without normal lube oil pressure while the pump is initially coming up to speed.
2. The makeup pumps perform as emergency high pressure injection pumps for LOCA.

Unavailability of system per WASH 1400:* 8.6×10^{-3}

Unavailability of component per WASH 1400:* pump, failure to start: 1×10^{-3}/D

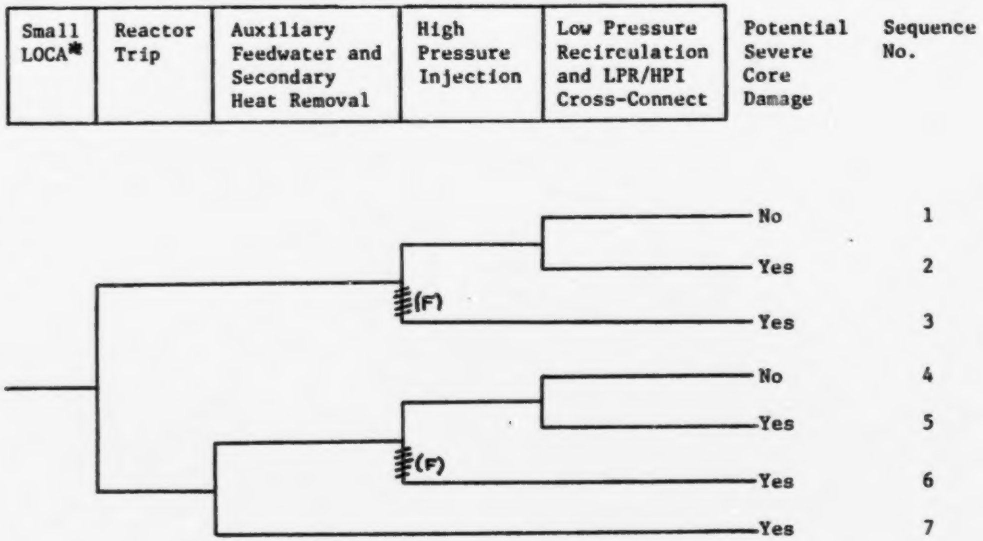
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Makeup pumps MU-PIA, MU-PIC running and voltage on buses for MU-PIA, MU-PIC interrupted for testing during refueling
MU-PIA, MU-PIC tripped
Power restored to MU-PIA, MU-PIC buses
MU-PIA start and run
MU-PIC start and run
Potential Severe Core Damage
Yes
No
No
No
No
No — No LOCA & unit in shutdown

NSIC 137384 — Actual Occurrence for Failure of Two Makeup Pumps During Testing at Three Mile Island 1

BEST DOCUMENT AVAILABLE


Small LOCA*
Reactor Trip
Auxiliary Feedwater and Secondary Heat Removal
High Pressure Injection
Low Pressure Recirculation and LPR/HPI Cross-Connect
Potential Severe Core Damage
Sequence No.
No 1
Yes 2
(F)
Yes 3
No 4
Yes 5
(F)
Yes 6
Yes 7

NSIC 137384 — Sequence of Interest for Failure of Two Makeup Pumps During Testing at Three Mile Island 1

*With loss of offsite power.

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 137384

DATE OF LER: April 5, 1978

DATE OF EVENT: March 18, 1978

SYSTEM INVOLVED: Makeup and High Pressure Injection

COMPONENT INVOLVED: Pumps

CAUSE: Malfunction of time delay relay.

SEQUENCE OF INTEREST: Small LOCA

ACTUAL OCCURRENCE: Relay failure.

REACTOR NAME: Three Mile Island 1

DOCKET NUMBER: 50-289

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 819 MWe

REACTOR AGE: 3.8 yr

VENDOR: B&W

ARCHITECT-ENGINEERS: Gilbert Associates, Inc.

OPERATORS: Metropolitan Edison

LOCATION: 10 miles SE of Harrisburg, PA

DURATION: 360(a) hours

PLANT OPERATING CONDITION: 0% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ________________

DISCOVERY METHOD: During test.

COMMENT: Lost 2 of 3 makeup/high pressure injection pumps because of faulty timers. These two pumps are normally those started by an SFAS signal.

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 137543

Date: April 13, 1978

Title: Diesel Generator Fails to Start Following a Loss of All Offsite Power at Calvert Cliffs 1

The failure sequence was:

1. With the plant shutdown, switchyard breaker protective relaying automatically opened the switchyard breakers, resulting in a loss of offsite power.
2. Diesel generator 11 failed to start.
3. Diesel generator 12 started and provided power to safety-related loads.

Corrective action:

None - the cause of the diesel-generator failing to start was not determined.

Design purpose of failed system or component:

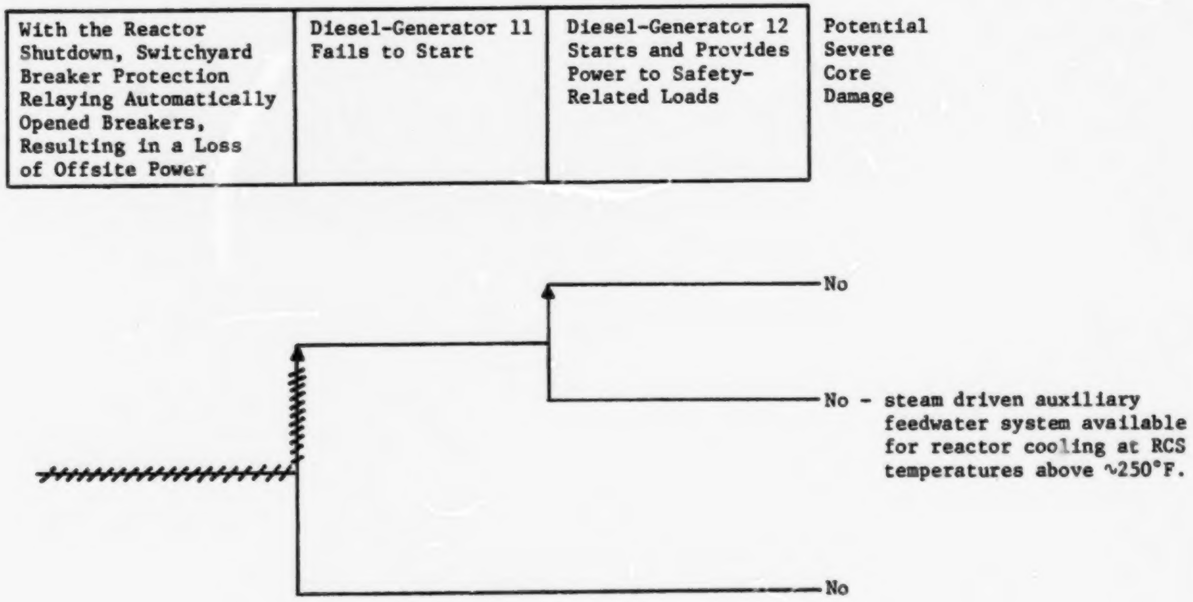
1. Offsite power provides the preferred source of power to safety-related equipment when the unit generator is not operating.
2. The diesel-generators provide a standby source of power to safety related loads when neither the unit generator nor offsite power is available.

Unavailability of system per WASH 1400:* offsite power: 2×10^{-5}/hr
onsite power: 1×10^{-2}/D

Unavailability of component per WASH 1400:* diesel generator: 3×10^{-2}/D

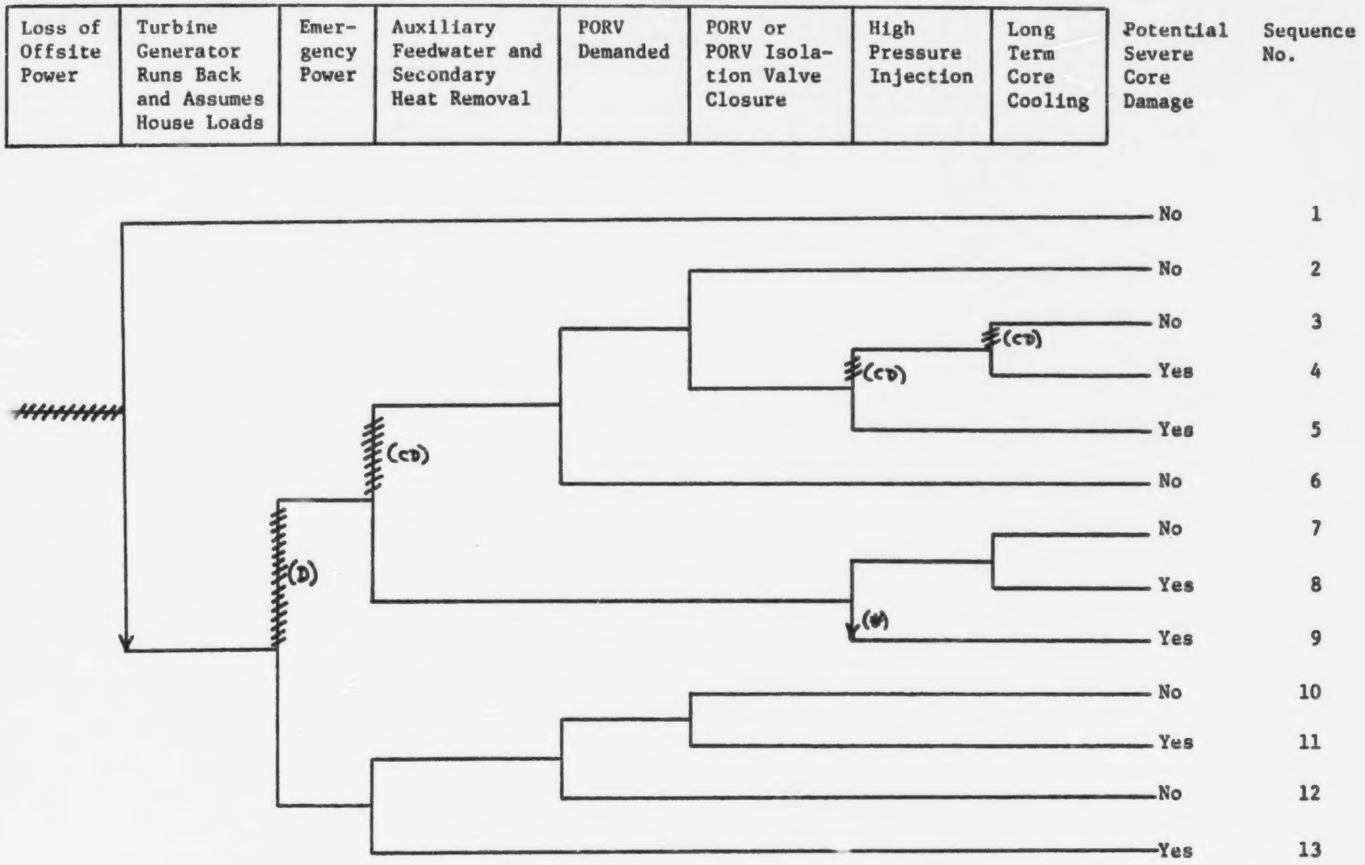
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


With the Reactor Shutdown, Switchyard Breaker Protection Relaying Automatically Opened Breakers, Resulting in a Loss of Offsite Power
Diesel-Generator 11 Fails to Start
Diesel-Generator 12 Starts and Provides Power to Safety-Related Loads
Potential Severe Core Damage
No
No - steam driven auxiliary feedwater system available for reactor cooling at RCS temperatures above ~250°F.
No

NSIC 137543 — Actual Occurrence for Diesel Generator Fails to Start Following a Loss of All Offsite Power at Calvert Cliffs 1

BEST DOCUMENT AVAILABLE


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
Yes 9
No 10
Yes 11
No 12
Yes 13
(CD)
(CD)
(CD)
(D)
(*)

NSIC 137543 — Sequence of Interest for Diesel Generator Fails to Start Following a Loss of All Offsite Power at Calvert Cliffs 1

*Not included in mitigation procedures.

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 137543

DATE OF LER: April 14, 1978

DATE OF EVENT: April 13, 1978

SYSTEM INVOLVED: offsite power, onsite power

COMPONENT INVOLVED: diesel generator

CAUSE: failure to start

SEQUENCE OF INTEREST: loss of offsite power

ACTUAL OCCURRENCE: loss of offsite power while shutdown

REACTOR NAME: Calvert Cliffs 1

DOCKET NUMBER: 50-317

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 845 MWe

REACTOR AGE: 3.5 yr

VENDOR: Combustion Engineering

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Baltimore Gas & Electric Co.

LOCATION: 40 miles south of Annapolis, Md.

DURATION: N/A

PLANT OPERATING CONDITION: shutdown

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; [✓ (b)] failed to start; (c) made inoperable; (d) ______________________

DISCOVERY METHOD: during operation

COMMENT: —

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 137918

Date: April 23, 1978

Title: Reactor Cooldown Exceeds Limit at Three Mile Island 2, Safety Injection at Three Mile Island 2

The failure sequence was:

1. A reactor trip occurred due to an instrumentation noise spike.
2. Following the turbine trip, four main steam relief valves on one SG and one on the other SG failed to reseat properly, eventually reseating about four minutes into the event.
3. The feedwater system was slow in isolating (2 1/2 minutes).
4. Safety injection occurred due to the depressurization of the RCS.
5. Pressurizer level was lost for approximately one minute.

Corrective action;

1. Prior to return to criticality, the relief valves were to be tested.
2. The source of the instrumentation noise spike was being investigated.
3. ICS tuning was to continue throughout initial plant startup to achieve proper transient response.

Design purpose of failed system or component:

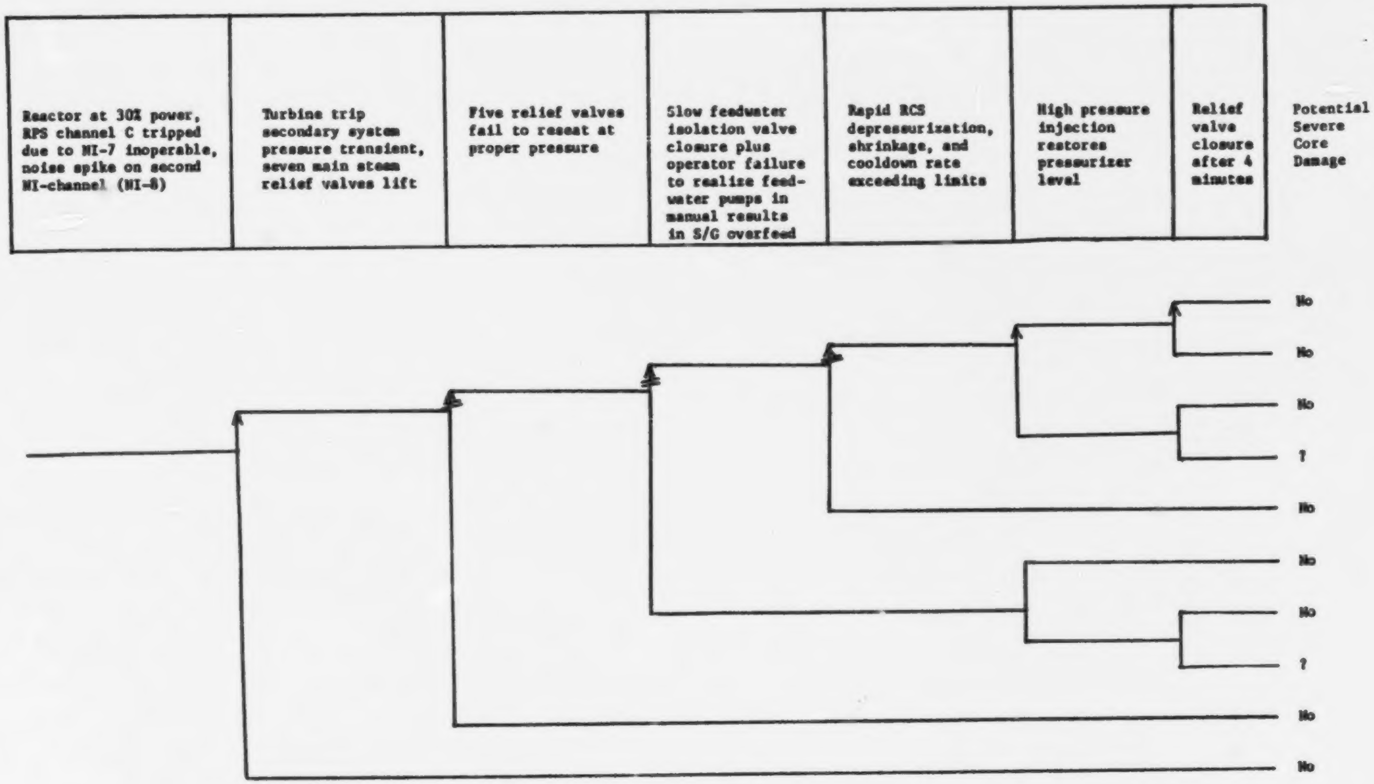
1. The relief valves provide over-pressure protection for the steam generators.
2. Feedwater isolation is provided to terminate flow to the steam generators in the event of a steam line break and other overcooling transients.

Unavailability of system per WASH 1400:* N/A

Unavailability of component per WASH 1400:* relief valve, failure to reseat, $\sim 10^{-2}$/D.

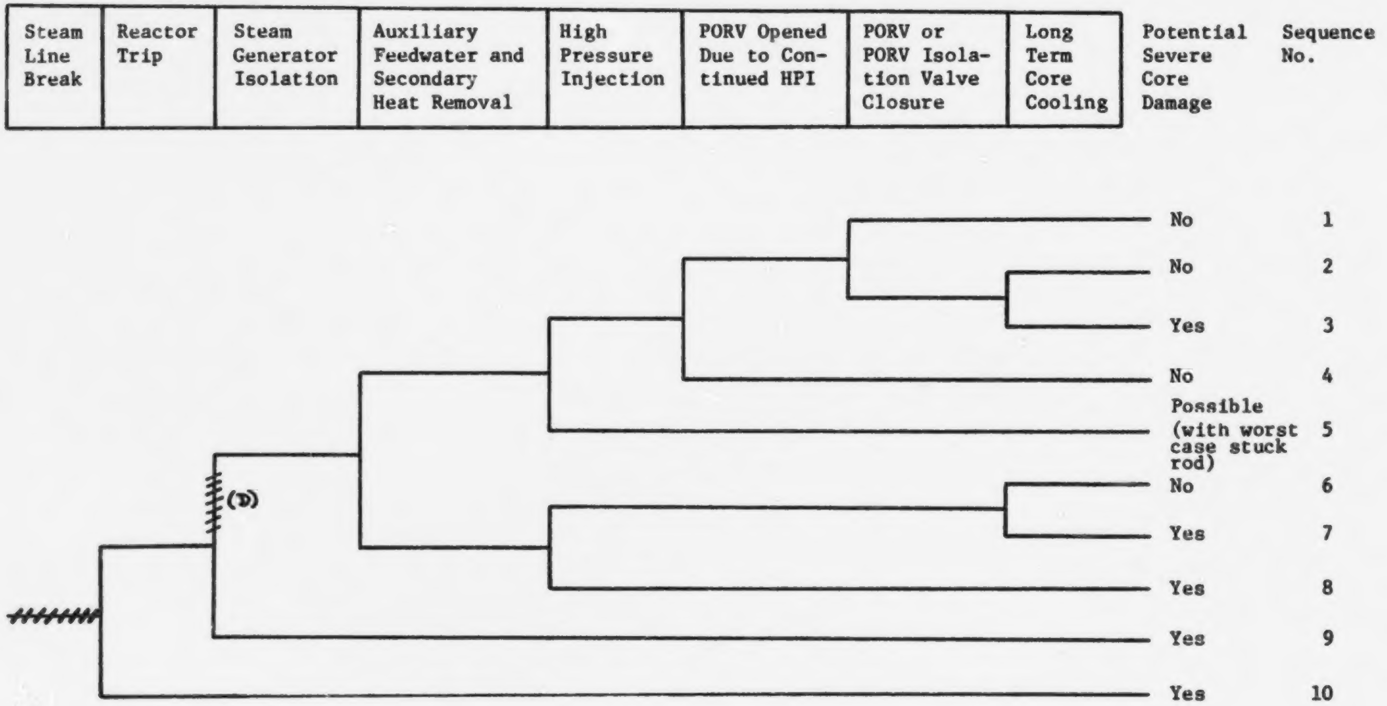
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor at 30% power, RPS channel C tripped due to NI-7 inoperable, noise spike on second NI-channel (NI-8)
Turbine trip secondary system pressure transient, seven main steam relief valves lift
Five relief valves fail to reseat at proper pressure
Slow feedwater isolation valve closure plus operator failure to realize feedwater pumps in manual results in S/G overfeed
Rapid RCS depressurization, shrinkage, and cooldown rate exceeding limits
High pressure injection restores pressurizer level
Relief valve closure after 4 minutes
Potential Severe Core Damage
No
No
No
?
No
No
No
?
No
No

NSIC 137918 — Actual Occurrence for Reactor Cooldown Exceeds Limit and Safety Injection at Three Mile Island 2

BEST DOCUMENT AVAILABLE


Steam Line Break
Reactor Trip
Steam Generator Isolation
Auxiliary Feedwater and Secondary Heat Removal
High Pressure Injection
PORV Opened Due to Continued HPI
PORV or PORV Isolation Valve Closure
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
(D)
No 1
No 2
Yes 3
No 4
Possible (with worst case stuck rod) 5
No 6
Yes 7
Yes 8
Yes 9
Yes 10

NSIC 137918 — Sequence of Interest for Reactor Cooldown Exceeds Limit and Safety Injection at Three Mile Island 2

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 137918

DATE OF LER: May 8, 1978

DATE OF EVENT: April 23, 1978

SYSTEM INVOLVED: Main steam, main feedwater

COMPONENT INVOLVED: Main steam relief valves

CAUSE: Relief valve failure to properly reseat, failure to properly secure feedwater.

SEQUENCE OF INTEREST: Steam line break

ACTUAL OCCURRENCE: Multiple stuck-open relief valves (long period blowdown)

REACTOR NAME: Three Mile Island 2

DOCKET NUMBER: 50-320

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 906 MWe

REACTOR AGE: .1 yr.

VENDOR: Babcock & Wilcox

ARCHITECT-ENGINEERS: Burns & Roe

OPERATORS: Metropolitan Edison

LOCATION: 10 miles SE of Harrisburg, PA

DURATION: N/A

PLANT OPERATING CONDITION: at 30% power

SAFETY FEATURE TYPE OF FAILURE: [✓ (a)] inadequate performance; (b) failed to start; (c) made inoperable; (d) ________________

DISCOVERY METHOD: transient following reactor trip

COMMENT: –

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 138830

Date: March 31, 1978

Title: Reactor Cooldown Rate Exceeds Limit Following Reactor Trip at Rancho Seco

The failure sequence was:

1. With the reactor at 70% power, an operator dropped a light bulb into an open back-lighted push button assembly while replacing a burned out light.
2. The dropped bulb created a short-to-ground in the Y portion of the Non-Nuclear Instrumentation (NNI-Y), which resulted in actuation of current limiting and under-voltage protection circuitry and consequently interrupted AC power to all NNI-Y DC power supplies.
3. Approximately 2/3 of all NNI signals were affected by the power loss, resulting in erroneous signals in both the control room and the Integrated Control System (ICS).
4. The ICS reduced feedwater flow to zero in response to faulty signals and reactor trip occurred on high pressure. An auxiliary feed pump started on the loss of feedwater flow, however the auxiliary feedwater valves remained closed due to incorrect steam generator level indication. (During the nine minute period following trip, both steam generators went dry.)

cont.

Corrective action:

1. An analysis of the effect of the transient on plant components was performed by B & W. As a result of this analysis, B & W recommended the plant be allowed to return to power, subject to increased surveillance and maneuvering limits for the first startup.
2. The steam lines were checked for damage.
3. An RCS leak test was performed.
4. A casualty procedure concerning restoration of NNI power was written, and a procedure for operator action in the event NNI was unavailable was to be prepared.

Design purpose of failed system or component:

The NNI provides control and instrumentation signals for numerous reactor plant parameters.

Unavailability of system per WASH 1400:* Not considered

Unavailability of component per WASH 1400:* —

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

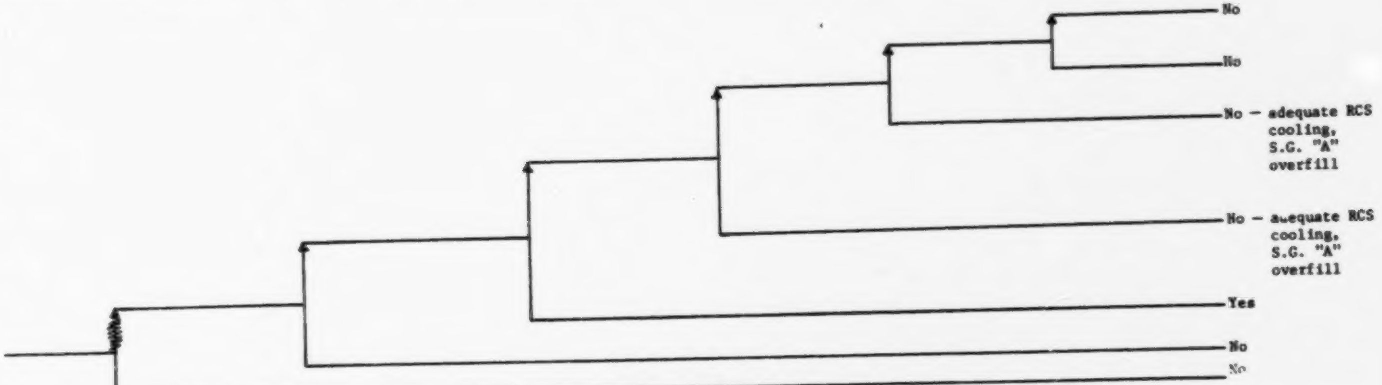
failure sequence, continued

5. The startup level indication for the "A" steam generator eventually (9 minutes after trip) drifted below the low level setpoint and the ICS opened the auxiliary feedwater control valve, admitting water to steam generator "A".

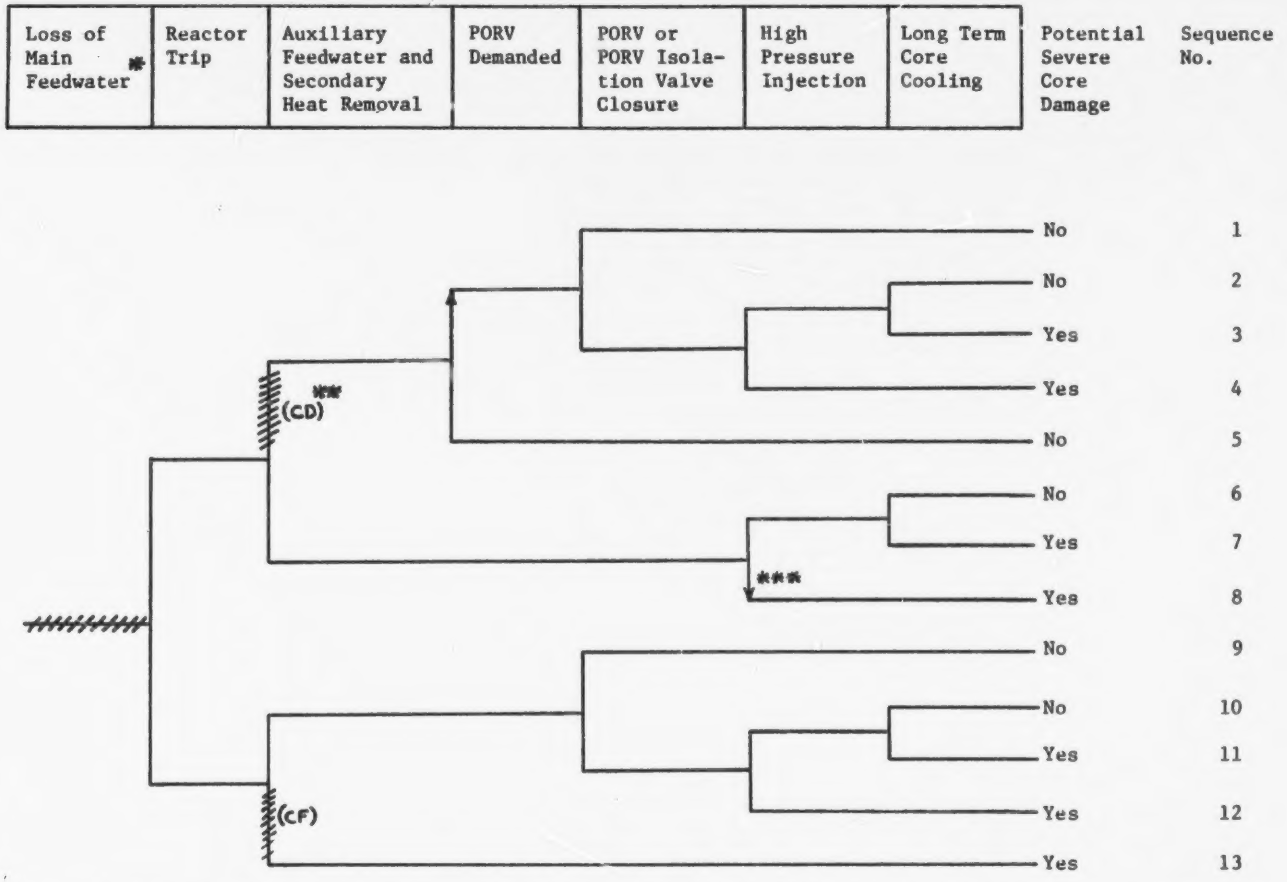
6. The injection of auxiliary feedwater to the "A" steam generator produced a drop in RCS pressure to below the SFAS low pressure trip setpoint (1600 psig), and resulted in the opening of both auxiliary feedwater bypass valves and the filling of both steam generators with water.

7. Due to the unavailability of reliable instrumentation, auxiliary feedwater injection was continued until NNI-Y power was restored (70 minutes after trip). RCS pressure was controlled using available pressurizer level and pressure instrumentation and throttling HPI flow. At the time NNI-Y power was restored, the steam generators had been filled; water had entered the main steam lines, and the RCS had cooled down to 285°F (300°F/hr cooldown rate).

Reactor at 70% power	Momentary short circuit results in trip of NNI-Y power supplies - 2/3 of all NNI control and instrumentation signals affected	ICS reduces feedwater flow to zero and starts auxiliary feedwater pump, flow to steam generator inhibited (all due to affected instrumentation), resulting in steam generator boiling dry. (reactor trip on high RCS pressure)	Startup level indication for "A" steam generator eventually drifts below low level setpoint - auxiliary feedwater flow initiated to "A" steam generator	Injection of auxiliary feedwater to "A" steam generator results in RCS cooldown - pressure drop below SFAS low pressure setpoint (1600 psig)	SFAS initiation provides AFW flow to second steam generator, high pressure injection to RCS	Due to lack of instrumentation, operators continue AFW injection until NNI-Y power is returned (70 minutes) and control RCS pressure and level by throttling HPI flow-steam generators filled, water in steam lines, RCS cooldown at 300°F/hr	Potential Severe Core Damage


No
No
No — adequate RCS cooling, S.G. "A" overfill
No — adequate RCS cooling, S.G. "A" overfill
Yes
No
No

NSIC 138830 — Actual Occurrence for Reactor Cooldown Rate Exceeds Limit Following Reactor Trip at Rancho Seco


Loss of Main Feedwater *
Reactor Trip
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
**
(CD)
***
(CF)
No 1
No 2
Yes 3
Yes 4
No 5
No 6
Yes 7
Yes 8
No 9
No 10
Yes 11
Yes 12
Yes 13

NSIC 138830 — Sequence of Interest for Reactor Cooldown Rate Exceeds Limit Following Reactor Trip at Rancho Seco

[*] due to NNI-Y power supply trip

[**] success requires instrument drift in correct direction or operator decision to open AFW control valves when instrumentation indicates steam generators contain water

[***] not part of mitigation procedure (pre-TMI-2)

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 138830

DATE OF LER: March 31, 1978

DATE OF EVENT: March 20, 1978

SYSTEM INVOLVED: Non Nuclear Instrumentation

COMPONENT INVOLVED: Power Supplies

CAUSE: Power supply trip due to momentary short, human error

SEQUENCE OF INTEREST: Loss of Main Feedwater

ACTUAL OCCURRENCE: Failure of NNI and consequent steam generator dryout and overfill, RCS overcooling.

REACTOR NAME: Rancho Seco

DOCKET NUMBER: 50-312

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 918 MWe

REACTOR AGE: 3.5 yr

VENDOR: B & W

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Sacramento Municipal Utility District

LOCATION: 25 mi SE of Sacramento, California

DURATION: N/A

PLANT OPERATING CONDITION: 70% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________________

DISCOVERY METHOD: operational event

COMMENT:

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 139565

Date: May 14, 1978

Title: Off-site Power Lost at St. Lucie 1

The failure sequence was:

1. With the reactor shut down for refueling, improper switching at one substation and improper connection of both 240 KV/138 KV autotransformer polarizing current transformer circuits at another substation resulted in a total loss of off-site power.
2. One diesel-generator was inoperative because of maintenance.
3. The second diesel-generator started and assumed safety-related loads until off-site power was restored.

Corrective action:

1. The improper polarization of circuits at the one substation was corrected.

Design purpose of failed system or component:

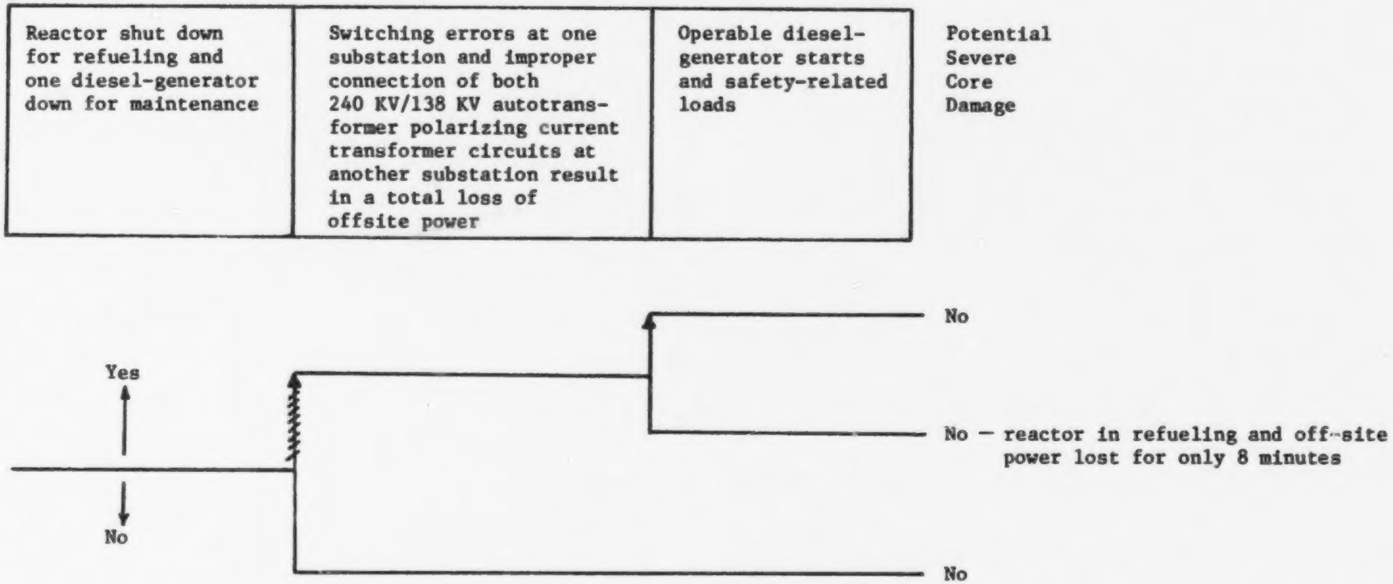
1. Off-site power provides the preferred source of electric power to safety-related equipment when the unit generator is not in operation.

Unavailability of system per WASH 1400:* loss of off-site power: 2×10^{-5}/hr

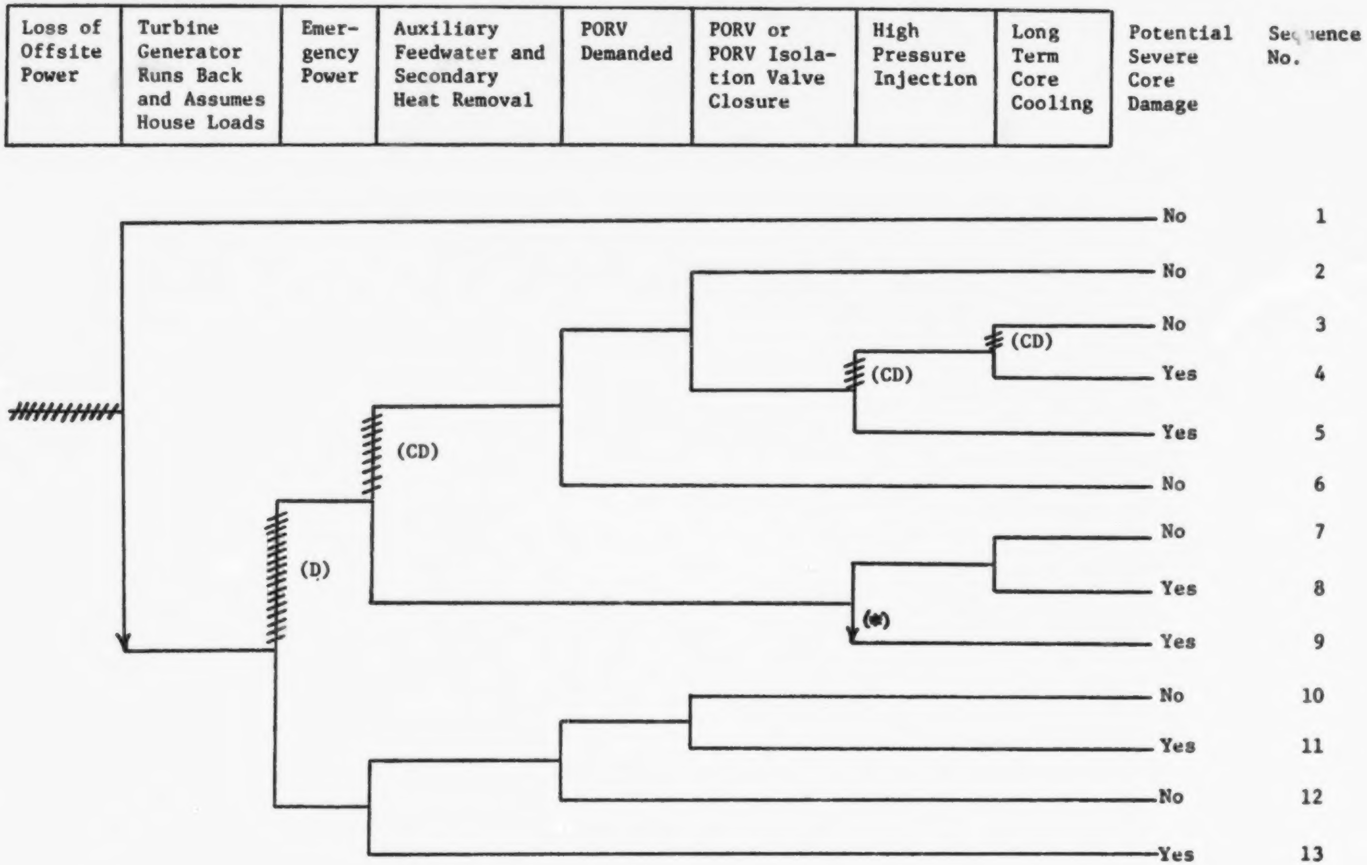
Unavailability of component per WASH 1400:* diesel generator: 3×10^{-2}/D

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor shut down for refueling and one diesel-generator down for maintenance
Switching errors at one substation and improper connection of both 240 KV/138 KV autotransformer polarizing current transformer circuits at another substation result in a total loss of offsite power
Operable diesel-generator starts and safety-related loads
Potential Severe Core Damage
Yes
No
No
No — reactor in refueling and off-site power lost for only 8 minutes
No

NSIC 139565 — Actual Occurrence for Offsite Power Lost at St. Lucie 1


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
(D)
(CD)
(CD)
(CD)
(*)
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
Yes 9
No 10
Yes 11
No 12
Yes 13

NSIC 139565 — Sequence of Interest for Offsite Power Lost at St. Lucie 1

*Not included in mitigation procedures.

BEST DOCUMENT AVAILABLE

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 139565

DATE OF LER: June 13, 1978

DATE OF EVENT: May 14, 1978

SYSTEM INVOLVED: off-site power

COMPONENT INVOLVED: switchyard switches and polarizing current transformer circuits

CAUSE: switching error and installation error, human error

SEQUENCE OF INTEREST: loss of offsite power with reactor trip

ACTUAL OCCURRENCE: loss of offsite power during refueling

REACTOR NAME: St. Lucie 1

DOCKET NUMBER: 50-335

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 802 MWe

REACTOR AGE: 2.1 yr

VENDOR: Combustion Engineering

ARCHITECT-ENGINEERS: Ebasco

OPERATORS: Florida Power & Light Co.

LOCATION: 12 miles SE of Fort Pierce, Fla.

DURATION: N/A

PLANT OPERATING CONDITION: refueling

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c) made inoperable]; (d) ________________

DISCOVERY METHOD: during operational event

COMMENT: –

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 139573

Date: June 29, 1978

Title: Both Diesel Generators Inoperable at Cook 2

The failure sequence was:

1. 1 DG had a fuel leak and was placed in the inoperable condition with switch in control room.
2. Operator entered wrong DG room and closed the starting air pilot valves erroneously on the 2nd DG.

Corrective action:

Repairs to DG with fuel leak.

Opened start air valve on 2nd DG.

Design purpose of failed system or component:

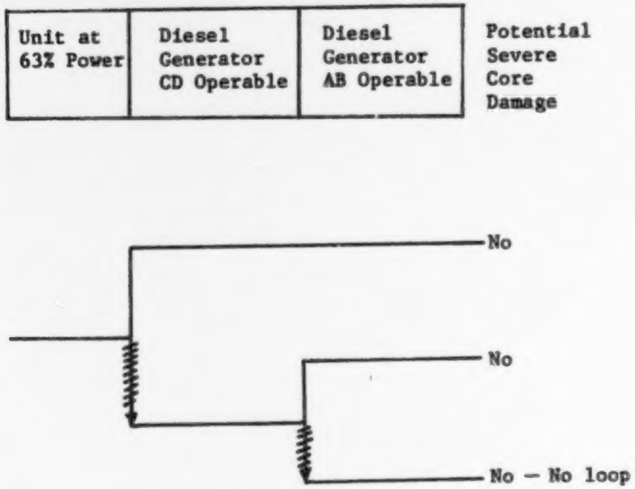
Provide ac power for ESF upon loss of normal and reserve ac power and to mitigate consequences of accidents such as LOCA.

Unavailability of system per WASH 1400:* emergency power: 10^{-2}/D

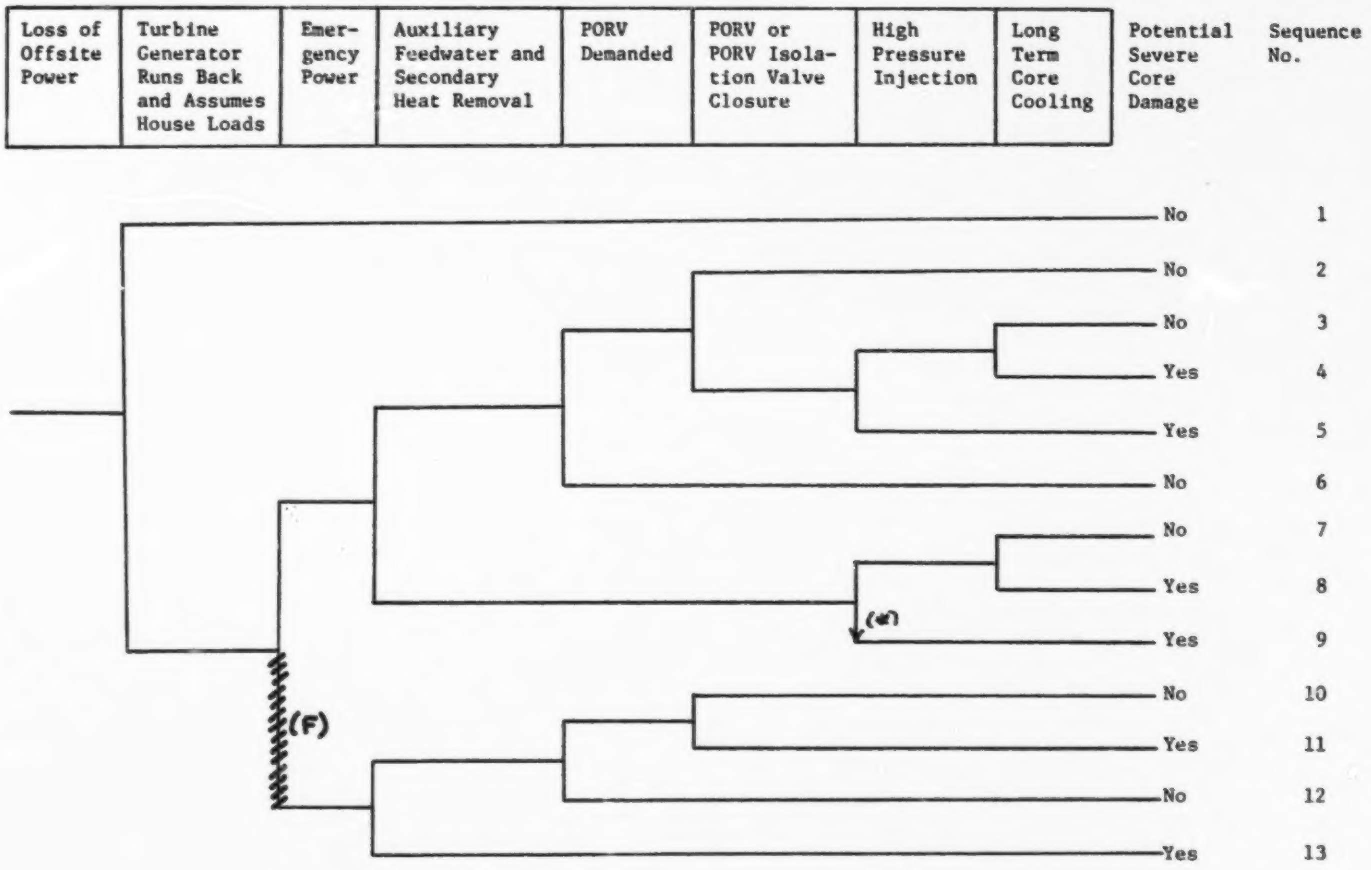
Unavailability of component per WASH 1400:* diesel generator: 3×10^{-2}/D

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Unit at 63% Power
Diesel Generator CD Operable
Diesel Generator AB Operable
Potential Severe Core Damage
No
No
No — No loop

NSIC 139573 — Actual Occurrence for Both Diesel Generators Inoperable at Cook 2


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
(F)
(*)
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
Yes 9
No 10
Yes 11
No 12
Yes 13

NSIC 139573 — Sequence of Interest for Both Diesel Generators Inoperable at Cook 2

*Not included in mitigation procedures.

BEST DOCUMENT AVAILABLE

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 139573

DATE OF LER: June 29, 1978

DATE OF EVENT: June 15, 1978

SYSTEM INVOLVED: Emergency on site power

COMPONENT INVOLVED: 2 diesel generators

CAUSE: Equipment failure + human error

SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE Equipment failure followed by operator error

REACTOR NAME: Cook 2

DOCKET NUMBER: 50-316

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 1100 MWe

REACTOR AGE: 0.2 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: American Electric Power

OPERATORS: Indiana & Michigan Electric Co.

LOCATION: 11 miles south of Benton Harbor, Mich.

DURATION: 2.9 hours

PLANT OPERATING CONDITION: 63% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ________________

DISCOVERY METHOD: Operational event

COMMENT: The layout (design) of the EDG rooms for the 2 unit plant was a factor in the operators error.

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: **140140**

Date: **August 6, 1978**

Title: **Loss of Offsite Power at Pilgrim 1**

The failure sequence was:

1. **The reactor was operating at 100% full power when lightning struck offsite distribution lines. This caused a ground fault in the high speed relays. This resulted in a loss of offsite power.**
2. **The reactor scrammed automatically.**
3. **The diesels started and loaded.**
4. **The RCIC/HPCI were manually started to maintain a proper water level.**
5. **Safety/relief valve 203-C was opened once for pressure control.**

Corrective action:

1. **Offsite power was returned to the startup transformers within 1/2 hour.**

Design purpose of failed system or component:

The transmission lines connect Pilgrim 1 with the New England Grid.

Unavailability of system per WASH 1400:* **loss of offsite power 2.0×10^{-5}/hr**

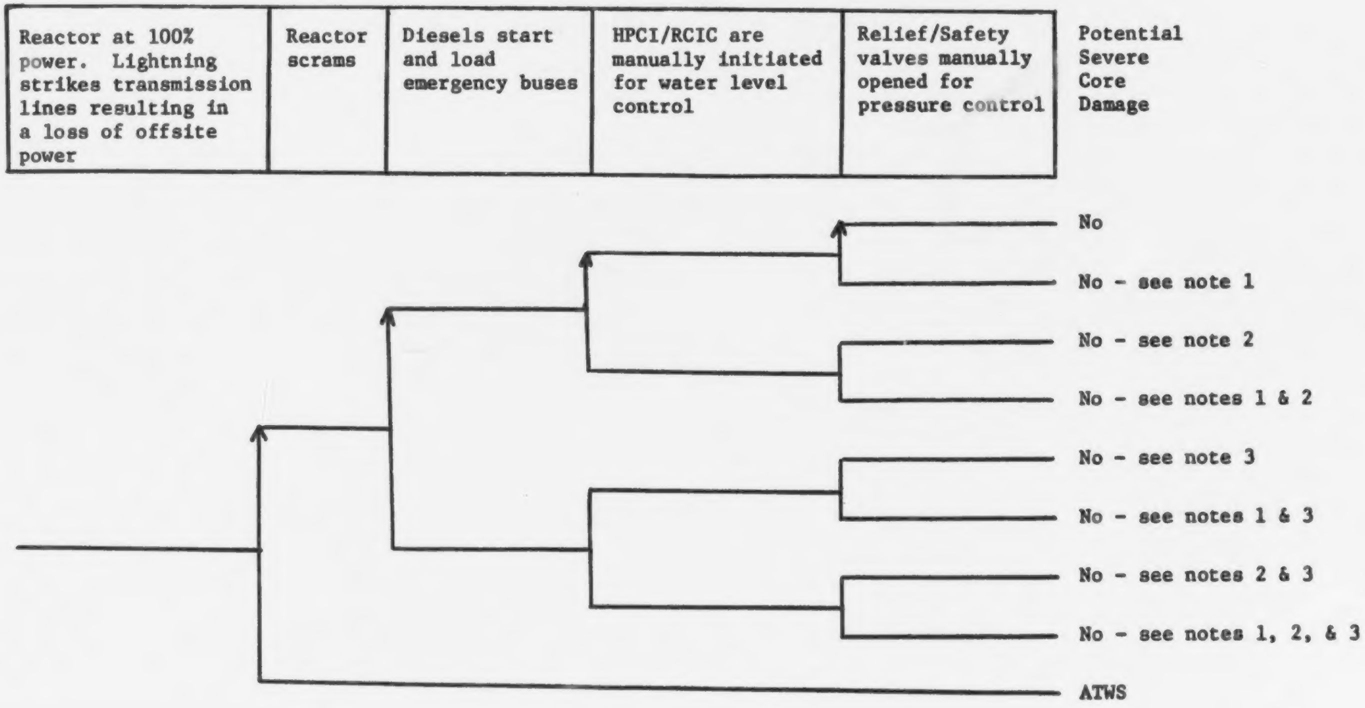
Unavailability of component per WASH 1400:* -

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

499

BEST DOCUMENT AVAILABLE


Reactor at 100% power. Lightning strikes transmission lines resulting in a loss of offsite power
Reactor scrams
Diesels start and load emergency buses
HPCI/RCIC are manually initiated for water level control
Relief/Safety valves manually opened for pressure control
Potential Severe Core Damage
No
No - see note 1
No - see note 2
No - see notes 1 & 2
No - see note 3
No - see notes 1 & 3
No - see notes 2 & 3
No - see notes 1, 2, & 3
ATWS

NSIC 140140 — Actual Occurrence for the Loss of Offsite Power at Pilgrim 1

[redacted]

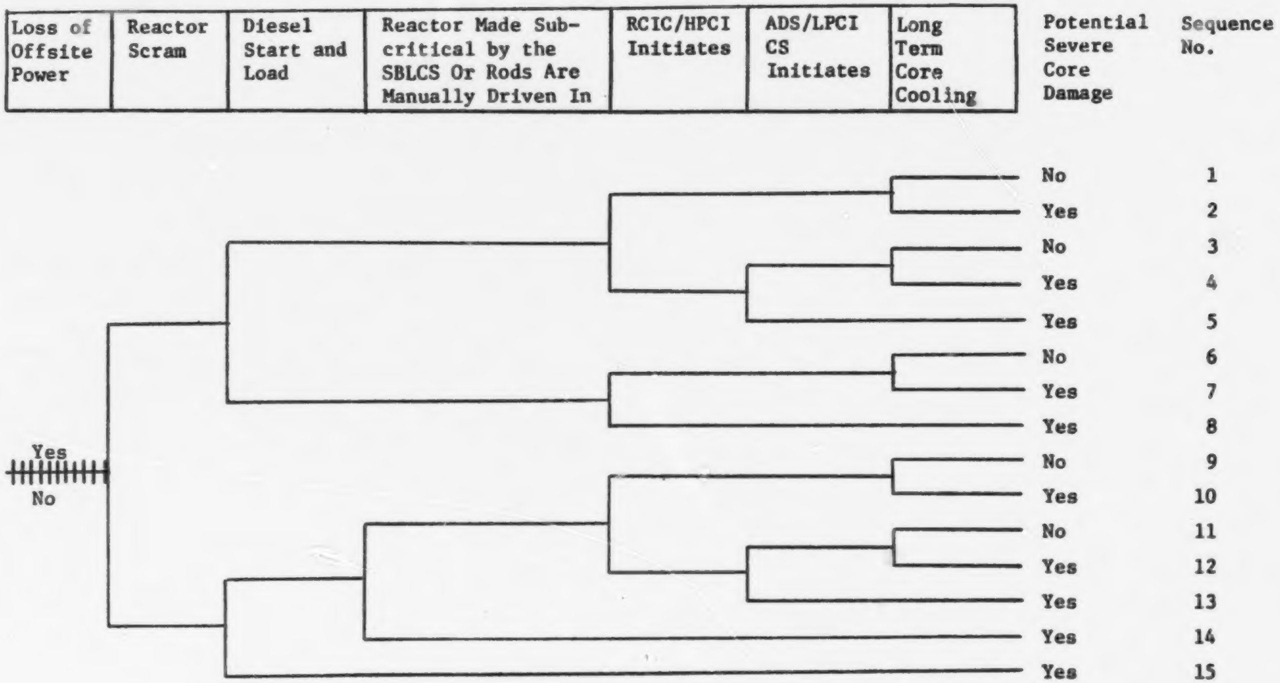
Notes

1. The safety and safety/relief valves automatically open at the following set points:

	No. of Valves	Set Pressure (PSIG)
Relief valves	3	1090 to 1240
Safety valves	2	1240

2. The HPCI and the RCIC autoactuate at either low-low reactor water inventory or high dry well pressure. See note 1.

3. The HPCI and RCIC are capable of initiation without an a.c. power source. The safety and relief valves also have this capability. The battery systems are capable of 8 hour operation without recharge. Since offsite power was restored within ∿½ hour, a potential core damage situation did not exist.


Loss of Offsite Power
Reactor Scram
Diesel Start and Load
Reactor Made Sub-critical by the SBLCS Or Rods Are Manually Driven In
RCIC/HPCI Initiates
ADS/LPCI CS Initiates
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
No 1
Yes 2
No 3
Yes 4
Yes 5
No 6
Yes 7
Yes 8
No 9
Yes 10
No 11
Yes 12
Yes 13
Yes 14
Yes 15

NSIC 140140 — Sequence of Interest for the Loss of Offsite Power at Pilgrim 1

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 140140

DATE OF LER: August 8, 1978

DATE OF EVENT: August 6, 1978

SYSTEM INVOLVED: electric power

COMPONENT INVOLVED: grounding relays

CAUSE: lightning striking offsite transmission line caused high speed relay operation for line to ground faults.

SEQUENCE OF INTEREST: loss of offsite power

ACTUAL OCCURRENCE: loss of offsite power

REACTOR NAME: Pilgrim 1

DOCKET NUMBER: 50-293

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 655 MWe

REACTOR AGE: 6.2 yr

VENDOR: GE

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Boston Edison

LOCATION: 4 miles SE of Plymouth, Mass.

DURATION: N/A

PLANT OPERATING CONDITION: 100% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ______________________

DISCOVERY METHOD: operational event

COMMENT: -

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 140162

Date: August 24, 1978

Title: Diesel Generator Fails to Start at Dresden 2

The failure sequence was:

1. Diesel generator 2/3 was out of service for maintenance.
2. Diesel generator 2 failed to start on the first attempt. The pinion gear on one of the air start motors did not engage with the diesel ring gear.

Corrective action:

The air starter motor was replaced and the diesel successfully started three times.

Design purpose of failed system or component:

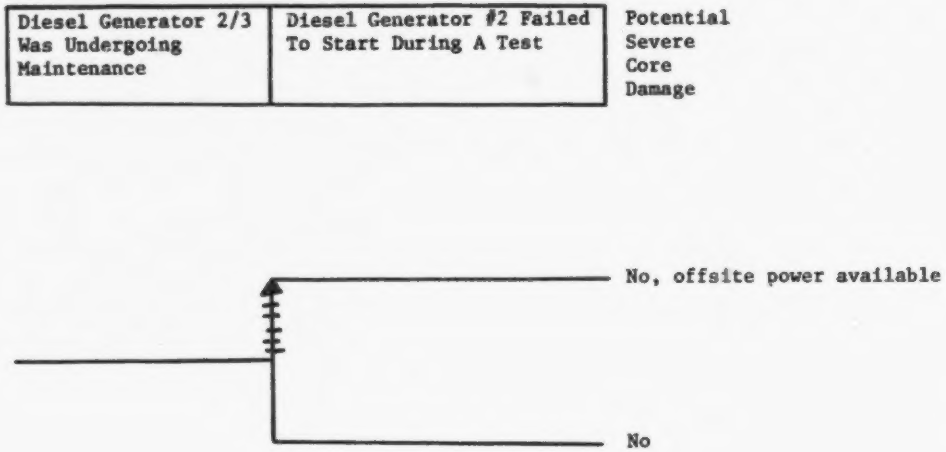
The emergency power system provides a.c. power when offsite sources are unavailable.

Unavailability of system per WASH 1400:* Emergency Power: 10^{-3}/D

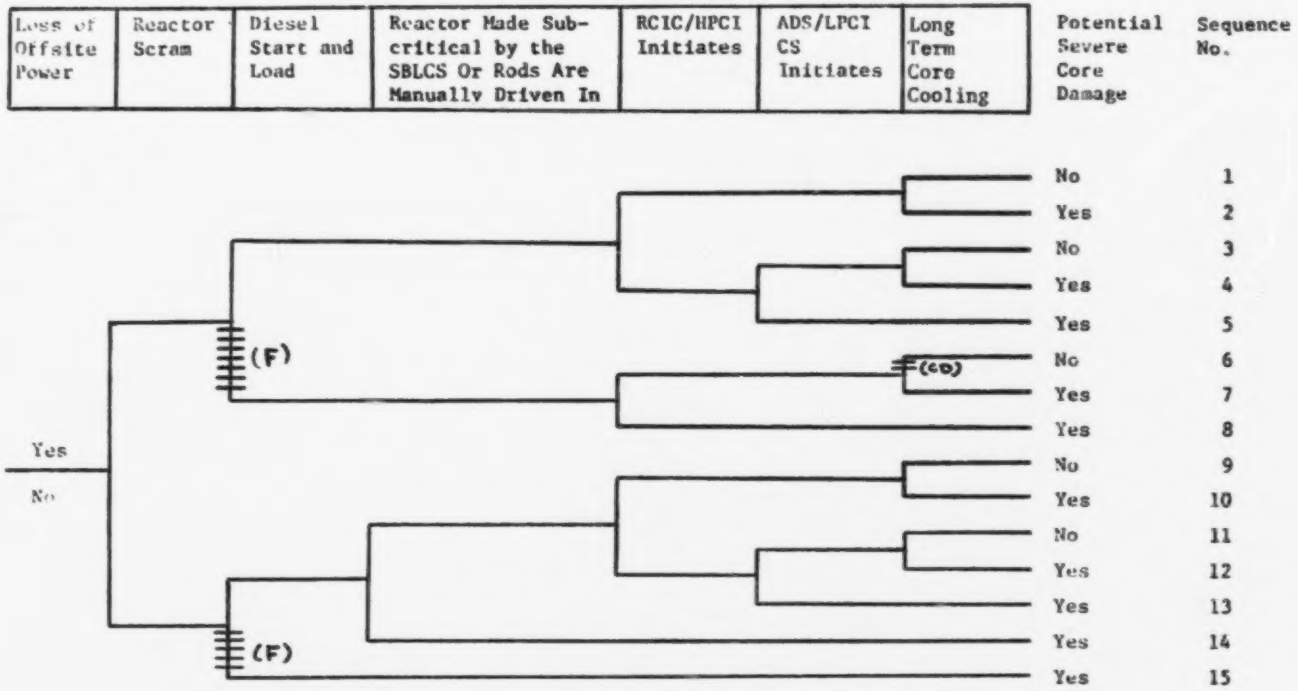
Unavailability of component per WASH 1400:* Diesel Generator failure to start: 3×10^{-2}/D

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Diesel Generator 2/3 Was Undergoing Maintenance
Diesel Generator #2 Failed To Start During A Test
Potential Severe Core Damage
No, offsite power available
No

NSIC 140162 — Actual Occurrence of Diesel Generator Fails to Start at Dresden 2


Loss of Offsite Power
Reactor Scram
Diesel Start and Load
Reactor Made Sub-critical by the SBLCS Or Rods Are Manually Driven In
RCIC/HPCI Initiates
ADS/LPCI CS Initiates
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
(F)
(F)
(co)
No 1
Yes 2
No 3
Yes 4
Yes 5
No 6
Yes 7
Yes 8
No 9
Yes 10
No 11
Yes 12
Yes 13
Yes 14
Yes 15

NSIC 140162 — Sequence of Interest for Diesel Generator Fails to Start at Dresden 2

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 140162

DATE OF LER: September 7, 1978

DATE OF EVENT: August 24, 1978

SYSTEM INVOLVED: Emergency Electric Power

COMPONENT INVOLVED: Diesel Generator Starting Motor

CAUSE: mechanical failure

SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE: Diesel Generator Fails to Start

REACTOR NAME: Dresden 2

DOCKET NUMBER: 50-237

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 794 MWe

REACTOR AGE: 8.6 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: Sargent & Lundy

OPERATORS: Commonwealth Edison

LOCATION: Nine miles E of Morris, Ill.

DURATION: 12 (a) hours

PLANT OPERATING CONDITION: 95% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________________

DISCOVERY METHOD: test

COMMENT:

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 140335

Date: August 11, 1978

Title: Loss of Offsite Power and One Diesel Generator at Beaver Valley 1

The failure sequence was:

1. With the reactor at 100% power, an electrical fault occurred in the main transformer.
2. All three reactor coolant pumps tripped on underfrequency, and a reactor trip, turbine trip, and safety injection occurred.
3. Offsite power was lost due to improper operation of the main generator out-of-step relay.
4. The No. 2 diesel generator failed to flash its field (no reason specified).

Corrective action:

The transformer was overhauled.

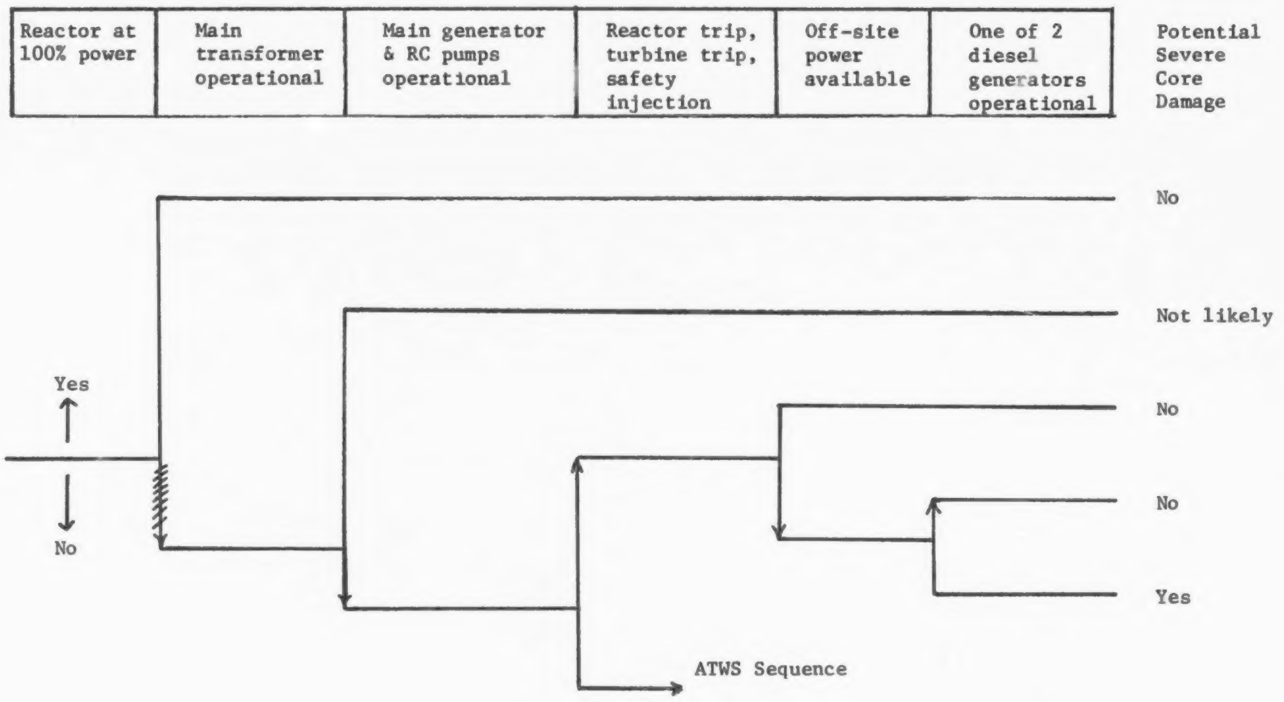
Design purpose of failed system or component:
To provide emergency power to ESF systems to mitigate the effects of accidents (LOCAs).

Unavailability of system per WASH 1400:* Loss of offsite power: 10^{-3}/D following trip

Unavailability of component per WASH 1400:* Diesel failure to start
$Q = 3.0 \times 10^{-2}/D$

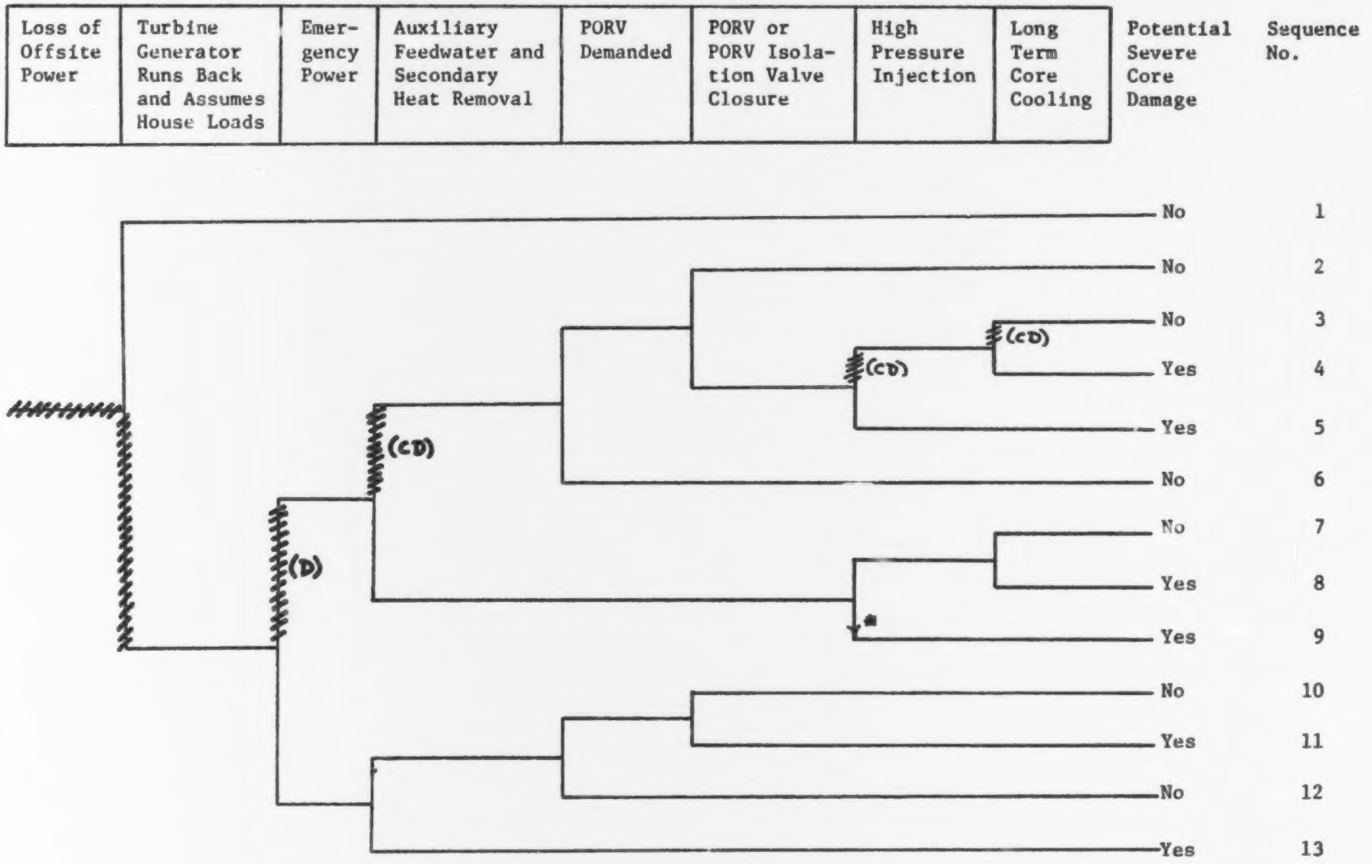
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor at 100% power
Main transformer operational
Main generator & RC pumps operational
Reactor trip, turbine trip, safety injection
Off-site power available
One of 2 diesel generators operational
Potential Severe Core Damage
No
Not likely
Yes
No
No
No
Yes
ATWS Sequence

NSIC 140335 — Actual Occurrence for Loss of Offsite Power and Diesel Generator Failure at Beaver Valley 1

BEST DOCUMENT AVAILABLE


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
(D)
(CD)
(CD)
(CD)
*
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
Yes 9
No 10
Yes 11
No 12
Yes 13

NSIC 140335 — Sequence of Interest for Loss of Offsite Power and Diesel Generator Failure at Beaver Valley 1

*not included in mitigation procedures

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 140335

DATE OF LER: August 11, 1978

DATE OF EVENT: July 28, 1978

SYSTEM INVOLVED: Off site power & onsite emergency power

COMPONENT INVOLVED: Main transformer and diesel generator

CAUSE: Electrical fault in station main transformer

SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE: Fault in main transformer

REACTOR NAME: Beaver Valley 1

DOCKET NUMBER: 50-334

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 852 MWe

REACTOR AGE: 2.2 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Stone & Webster

OPERATORS: Duquesne Light Co.

LOCATION: 5 miles east of E. Liverpool, Ohio

DURATION: N/A

PLANT OPERATING CONDITION: 100% power

SAFETY FEATURE TYPE OF FAILURE: [✓ (a)] inadequate performance; (b) failed to start;
(c) made inoperable; (d) ________________

DISCOVERY METHOD: Operational event

COMMENT: -

[redacted]

1

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 140691

Date: September 21, 1978

Title: Both Diesel Generators Out of Service at Beaver Valley 1

The failure sequence was:

1. With the reactor at 0% power, diesel generator No. 2 was removed from service for testing of the diesel air system and oil system relief valves.
2. During surveillance testing of diesel generator No. 1, the output breaker failed to close using the control switch. The breaker was manually closed and the diesel run for one hour at full load. The diesel was successfully synchronized three times using the control switch.

Corrective action:

Diesel generator No. 1 was subsequently tested and its output breaker performed satisfactorily. A test panel was installed to monitor the breaker closing contacts to detect abnormal operation.

Design purpose of failed system or component:

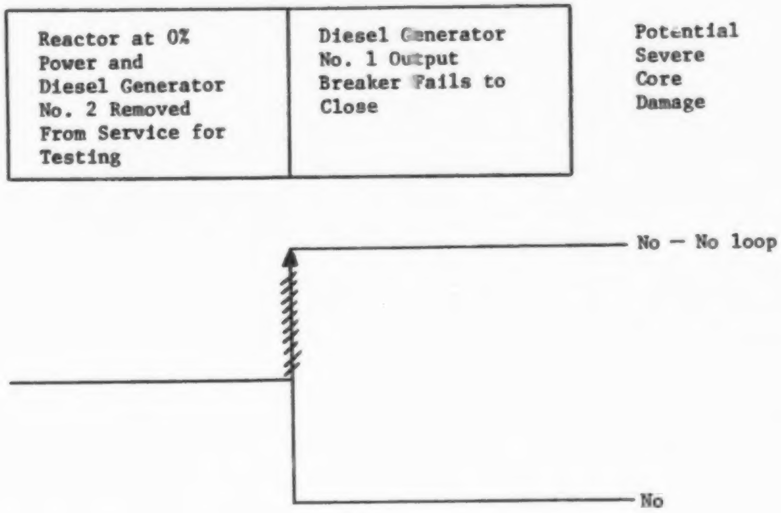
The diesel generators provide power to safety-related equipment when the unit generator and offsite power sources are unavailable.

Unavailability of system per WASH 1400:* Emergency power: $1 \times 10^{-2}/D$

Unavailability of component per WASH 1400:* Diesel generators: $3 \times 10^{-2}/D$

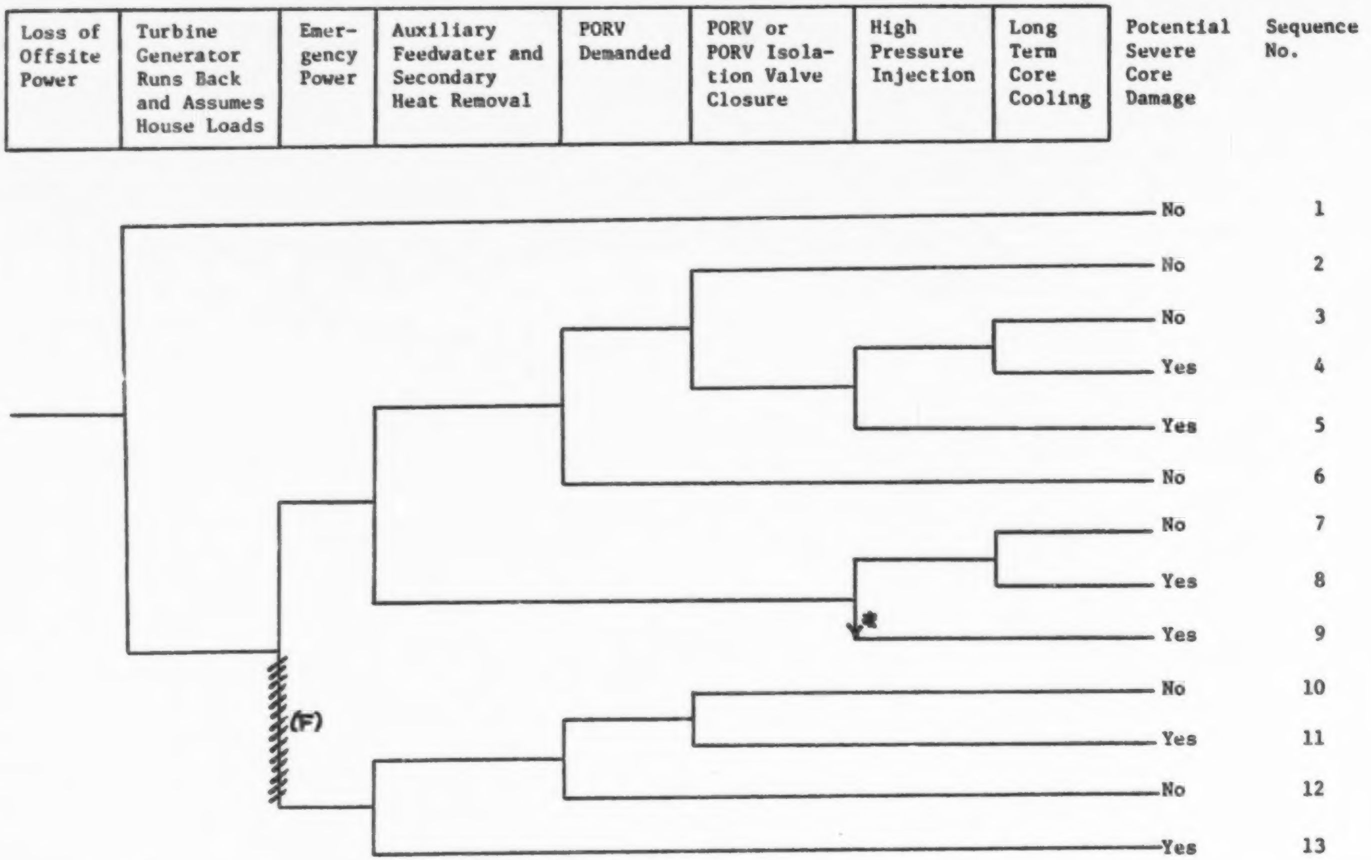
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor at 0% Power and Diesel Generator No. 2 Removed From Service for Testing
Diesel Generator No. 1 Output Breaker Fails to Close
Potential Severe Core Damage
No — No loop
No

NSIC 140691 — Actual Occurrence for Two Diesel Generators Unavailable at Beaver Valley 1

BEST DOCUMENT AVAILABLE


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
Yes 9
No 10
Yes 11
No 12
Yes 13
(F)
*

NSIC 140691 — Sequence of Interest for Two Diesel Generators Unavailable at Beaver Valley 1

*Not in operating procedures

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 140691

DATE OF LER: September 21, 1978

DATE OF EVENT: September 5, 1978

SYSTEM INVOLVED: Emergency electric power

COMPONENT INVOLVED: Diesel generators

CAUSE: Failure of output breaker to close on one diesel while other diesel was unavailable due to testing
SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE: Two diesel generators unavailable simultaneously

REACTOR NAME: Beaver Valley 1

DOCKET NUMBER: 334

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 852 MWe

REACTOR AGE: 2.3 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Stone and Webster

OPERATORS: Duquesne Light Company

LOCATION: 5 miles East of East Liverpool, Ohio

DURATION: 4.3 hours
PLANT OPERATING CONDITION: 0% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; [✓ (b)] failed to start; [✓ (c)] made inoperable; (d) ____________________

DISCOVERY METHOD: During testing

COMMENT:

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 140693

Date: September 26, 1978

Title: Both Diesel Generators Out of Service at Beaver Valley 1

The failure sequence was:

1. With the reactor at 0% power, diesel generator No. 2 was removed from service for testing of the diesel air system and oil system relief valves.
2. During surveillance testing of diesel generator No. 1, the output breaker failed to close using the control switch. During subsequent testing, the breaker closed on the first attempt.

Corrective action:

Diesel generator No. 1 was subsequently tested and its output breaker performed satisfactorily. A test panel was installed to monitor the breaker closing contacts to detect abnormal operation.

Design purpose of failed system or component:

The diesel generators provide power to safety-related equipment when the unit generator and offsite power sources are unavailable.

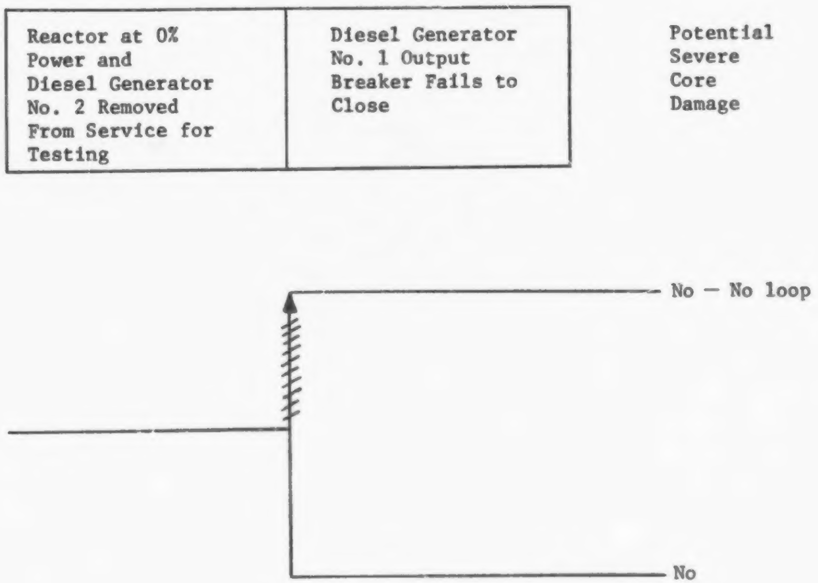
Unavailability of system per WASH 1400:* Emergency power: 1×10^{-2}/D

Unavailability of component per WASH 1400:* Diesel generators: 3×10^{-2}/D

---

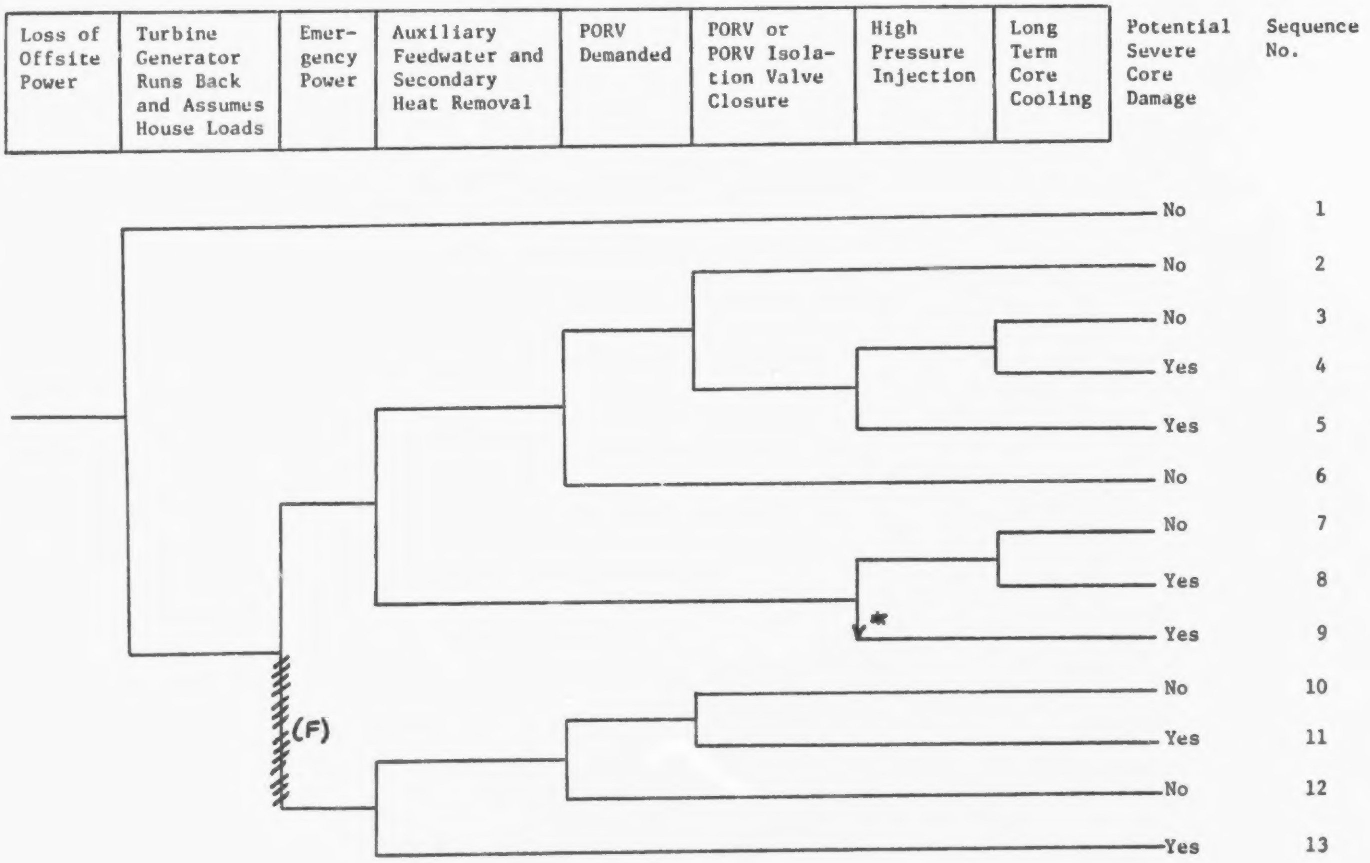
*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

516


Reactor at 0% Power and Diesel Generator No. 2 Removed From Service for Testing
Diesel Generator No. 1 Output Breaker Fails to Close
Potential Severe Core Damage
No — No loop
No

NSIC 140693 — Actual Occurrence for Both Diesel Generators Out of Service at Beaver Valley 1

BEST DOCUMENT AVAILABLE


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
Yes 9
No 10
Yes 11
No 12
Yes 13
(F)
*

NSIC 140693 — Sequence of Interest for Both Diesel Generators Out of Service at Beaver Valley 1

*Not included in operating procedures

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 140693

DATE OF LER: September 26, 1978

DATE OF EVENT: September 12, 1978

SYSTEM INVOLVED: Emergency electric power

COMPONENT INVOLVED: Diesel generators

CAUSE: Failure of output breaker to close on one diesel while other diesel was unavailable due to testing

SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE: Two diesel generators unavailable simultaneously

REACTOR NAME: Beaver Valley 1

DOCKET NUMBER: 334

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 852 MWe

REACTOR AGE: 2.3 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Stone and Webster

OPERATORS: Duquesne Light Company

LOCATION: 5 miles East of East Liverpool, Ohio

DURATION: .6 hours

PLANT OPERATING CONDITION: 0% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; [✓ (b)] failed to start; [✓ (c)] made inoperable; (d) ____________

DISCOVERY METHOD: During testing

COMMENT:

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 141097

Date: March 29, 1978

Title: Safety Injection Occurs at TMI-2

The failure sequence was:

1. Alternate power source feeder breakers to vital bus 2-IV (120 V a-c) were tripped for test.
2. Frequency control module failed resulting in blown fuses in the DC input to invertor.
3. Vital bus 2-IV tripped on invertor failure, resulting in the PORV failing open.
4. Because of the reactor coolant pump configuration at the time, a reactor trip also occurred.
5. The operators did not realize the relief valve had failed open because of lack of PORV position indication.
6. ECCS was initiated at 1650 psig.
7. Four minutes into the transient, vital bus 2-IV was re-energized, which closed the relief valve and terminated the transient.

Corrective action;

1. A design change was initiated to change the failure position of the relief valve upon loss of power to closed in lieu of open.
2. Relief valve position indication was provided in the control room.

Design purpose of failed system or component:

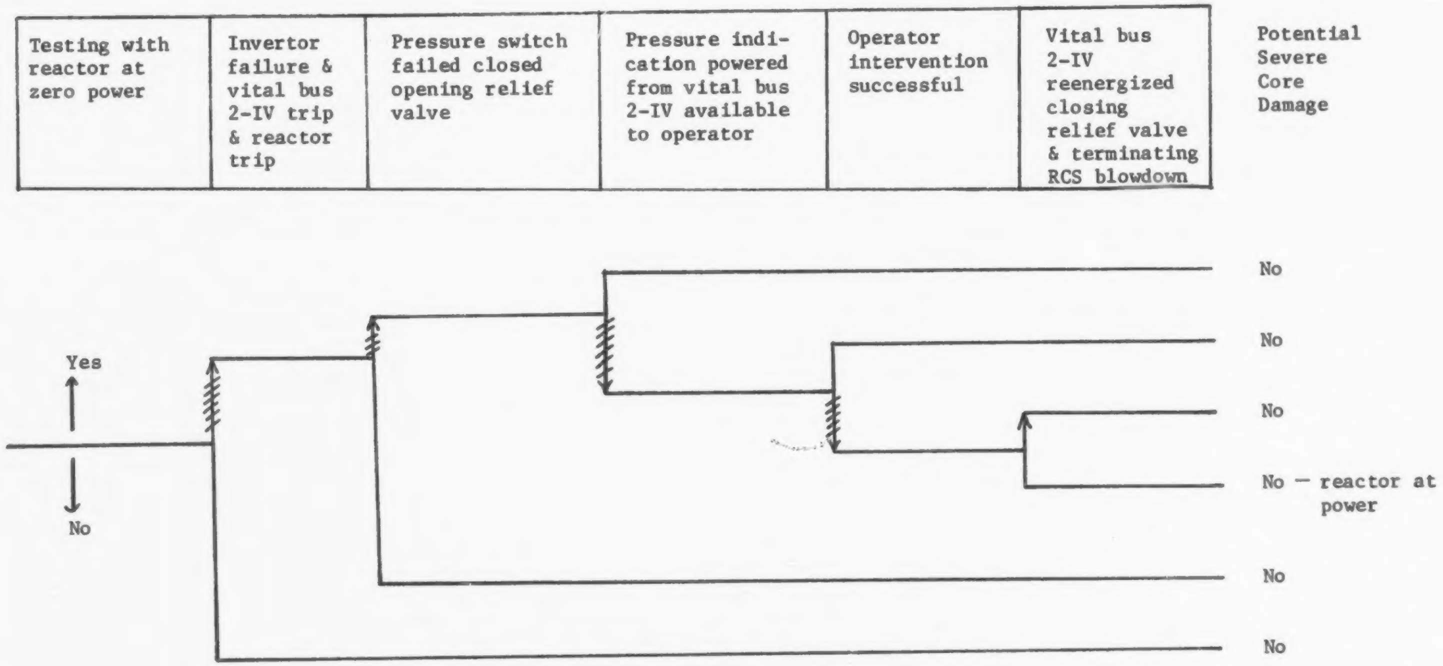
1. Provide 120 V a-c power for loads which require uninterrupted power.
2. The pilot operated relief valve provides overpressure protection for the RCS.

Unavailability of system per WASH 1400:* N/A

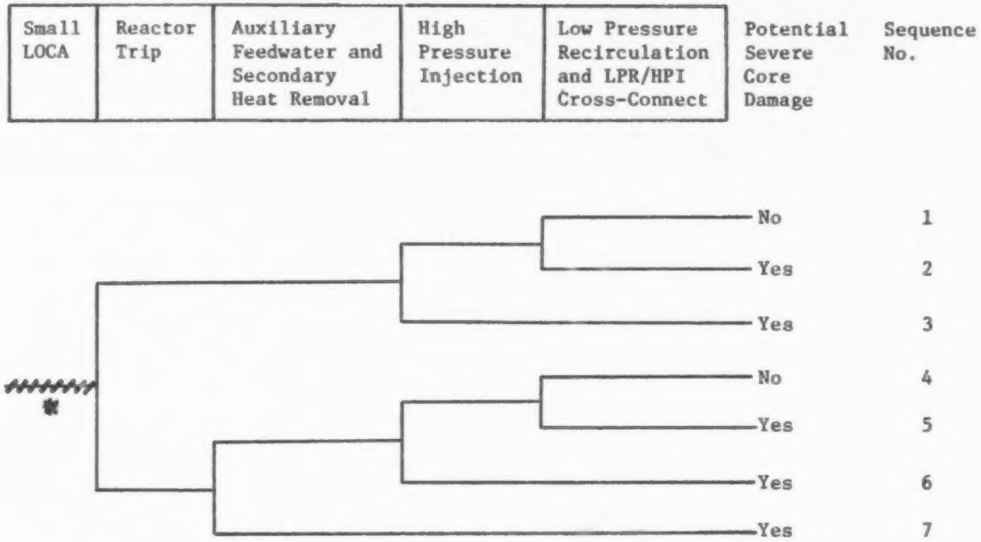
Unavailability of component per WASH 1400:* solid state device, high power application: 3×10^{-6}/hr.

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Testing with reactor at zero power
Invertor failure & vital bus 2-IV trip & reactor trip
Pressure switch failed closed opening relief valve
Pressure indication powered from vital bus 2-IV available to operator
Operator intervention successful
Vital bus 2-IV reenergized closing relief valve & terminating RCS blowdown
Potential Severe Core Damage
Yes
No
No
No
No
No — reactor at power
No
No

NSIC 141097 — Actual Occurrence for Loss of Vital Bus and Relief Valve Opening at Three Mile Island 2


Small LOCA
Reactor Trip
Auxiliary Feedwater and Secondary Heat Removal
High Pressure Injection
Low Pressure Recirculation and LPR/HPI Cross-Connect
Potential Severe Core Damage
Sequence No.
*
No 1
Yes 2
Yes 3
No 4
Yes 5
Yes 6
Yes 7

NSIC 141097 — Sequence of Interest for Safety Injection Occurs at Three Mile Island 2

*failure requires operator error of failing to close the PORV isolation valve.

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 141097

DATE OF LER: June 27, 1978

DATE OF EVENT: March 29, 1978

SYSTEM INVOLVED: 120 V a-c vital power, RCS

COMPONENT INVOLVED: Vital bus invertor

CAUSE: Blown fuse, faulty invertor control module

SEQUENCE OF INTEREST: small LOCA - stuck open PORV

ACTUAL OCCURRENCE: stuck open PORV

REACTOR NAME: Three Mile Island 2

DOCKET NUMBER: 50-320

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 906 MWe

REACTOR AGE: 0.0+ yr

VENDOR: Babcock & Wilcox

ARCHITECT-ENGINEERS: Burns & Roe

OPERATORS: Metropolitan Edison Co.

LOCATION: 10 miles SE of Harrisburg, PA

DURATION: N/A

PLANT OPERATING CONDITION: Zero power physics testing

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; (c) made inoperable; [✓ (d) failed in unsafe position]

DISCOVERY METHOD: incident during physics testing

COMMENT: Vital bus tripped because of invertor failure while alternate power source was removed for testing.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 141455

Date: October 4, 1978

Title: Both Diesel Generators Inoperable Simultaneously at Rancho Seco

The failure sequence was:

1. While one diesel generator was out of service for maintenance, the second diesel generator was started and paralleled with the nuclear service bus, as required.
2. While the diesel was running, fuel oil began to spray from a loose packing, which could not be tightened while the diesel was in operation.
3. The second diesel was shut down for repair.

Corrective action:

1. The running diesel was shut down to repair the packing gland. The reactor was shut down as required. The packing gland was repaired, the diesel generator restarted, and the plant returned to power.

Design purpose of failed system or component:

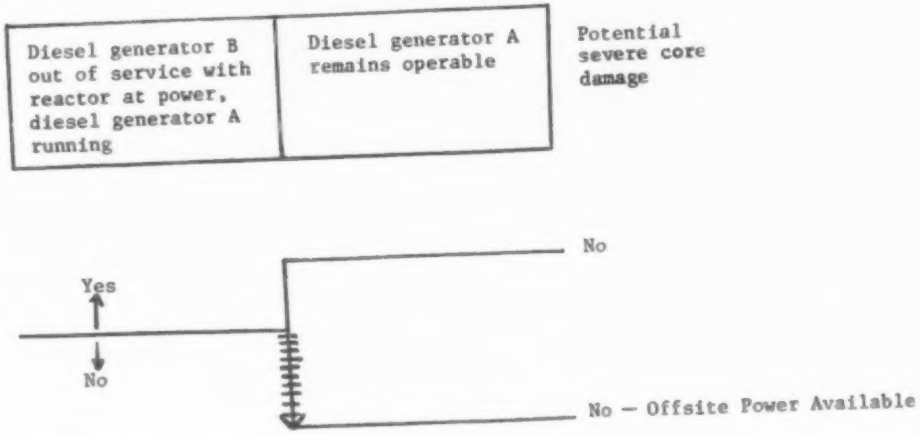
1. The failed component provides removal of foreign matter from diesel generator fuel oil.

Unavailability of system per WASH 1400:* emergency power: 10^{-2}/D

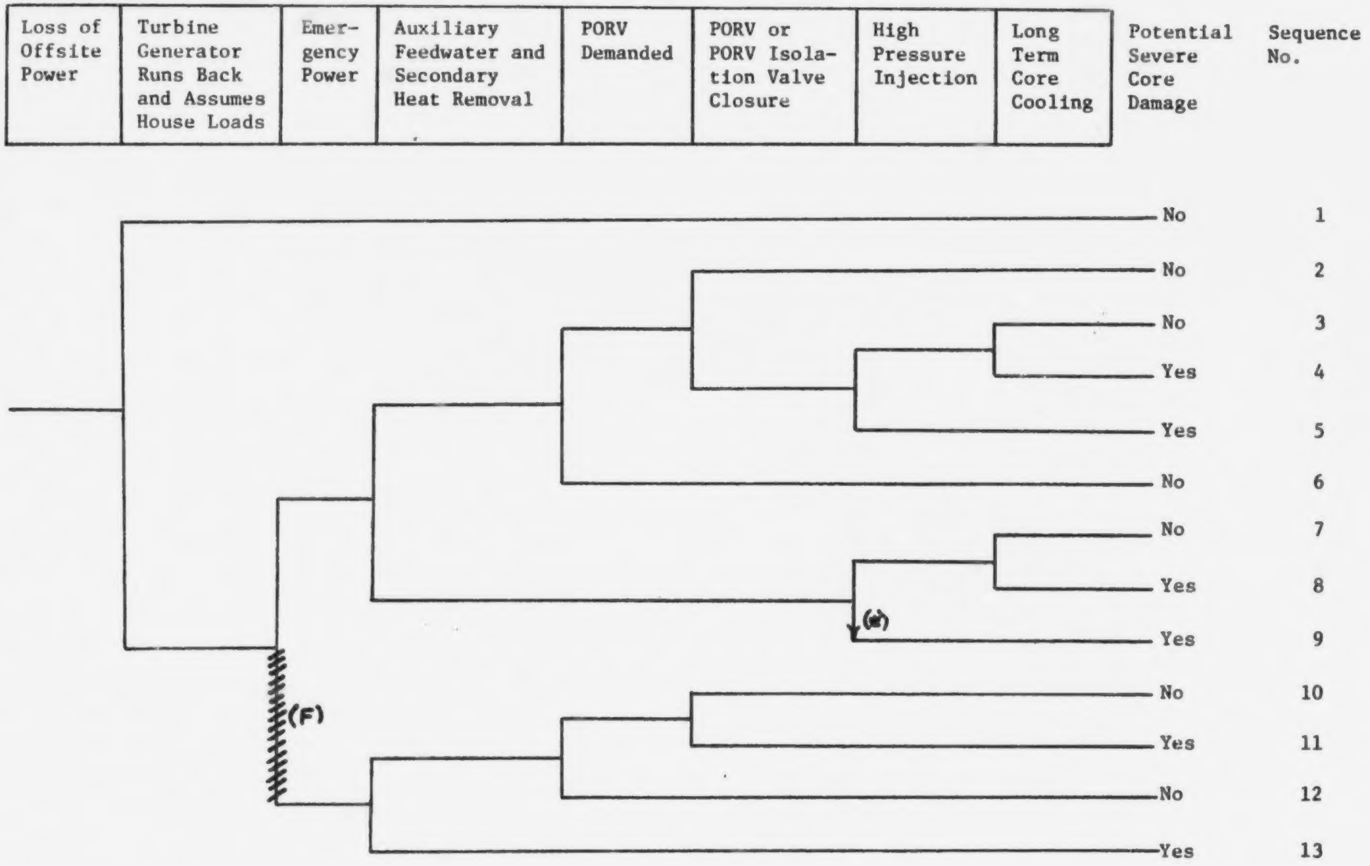
Unavailability of component per WASH 1400:* individual diesel generator: 3×10^{-2}/D

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Diesel generator B out of service with reactor at power, diesel generator A running
Diesel generator A remains operable
Potential severe core damage
Yes
No
No
No — Offsite Power Available

NSIC 141455 — Actual Sequence for Both Diesel Generators Inoperable Simultaneously at Rancho Seco


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
(*)
Yes 9
(F)
No 10
Yes 11
No 12
Yes 13

NSIC 141455 — Sequence of Interest for Both Diesel Generators Inoperable Simultaneously at Rancho Seco

*Not included in mitigation procedures.

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 141455

DATE OF LER: October 17, 1978

DATE OF EVENT: October 4, 1978

SYSTEM INVOLVED: Emergency electric power

COMPONENT INVOLVED: Packing gland — diesel generators

CAUSE: Loose packing gland on fuel oil strainer.

SEQUENCE OF INTEREST: Loss of offsite power.

ACTUAL OCCURRENCE: 1 DG out of service — loose gland resulted in oil spray of fuel oil from second

REACTOR NAME: Rancho Seco

DOCKET NUMBER: 50-312

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 918 MWe

REACTOR AGE: 4.2 yr

VENDOR: B&W

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Sacramento Municipal Utility District

LOCATION: 25 miles SE of Sacramento, Ca.

DURATION: .9 hours

PLANT OPERATING CONDITION: 100% power.

SAFETY FEATURE TYPE OF FAILURE: [✓ (a)] inadequate performance; (b) failed to start; (c) made inoperable; (d) ________________

DISCOVERY METHOD: Operational event

COMMENT: 1 DG was being serviced. Loose packing gland on fuel oil strainer caused fuel oil to spray out. Location of strainer prevented tightening gland with diesel running. DG was shut down to tighten gland.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 141523

Date: October 31, 1978

Title: Cavitation of both LPSI Pumps at Calvert Cliffs 2

The failure sequence was:

1. The LPSI pumps were operating in the shutdown cooling mode.
2. Purification system resin was transferred using air.
3. Air leaked into shutdown cooling system.
4. Both LPSI pumps became airbound and cavitated. (2 hours required to restore shutdown cooling.)

Corrective action:

Revised operating instructions to prohibit transferring resin while purification and shutdown cooling system are cross connected.

Design purpose of failed system or component:

Provide a large volume of water to the reactor pressure vessel during the early post-accident period fillowing a large LOCA.

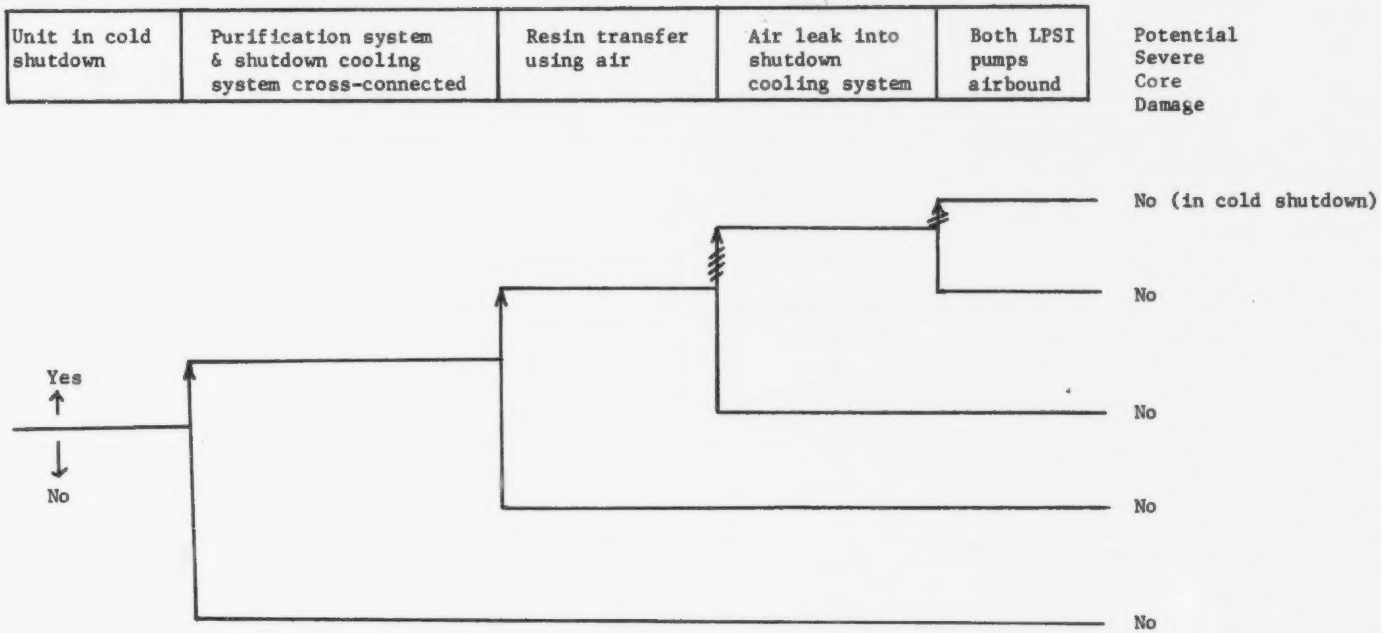
Unavailability of system per WASH 1400:* high pressure recirculation: 9×10^{-3}/D

Unavailability of component per WASH 1400:* pump: 1×10^{-3}/D

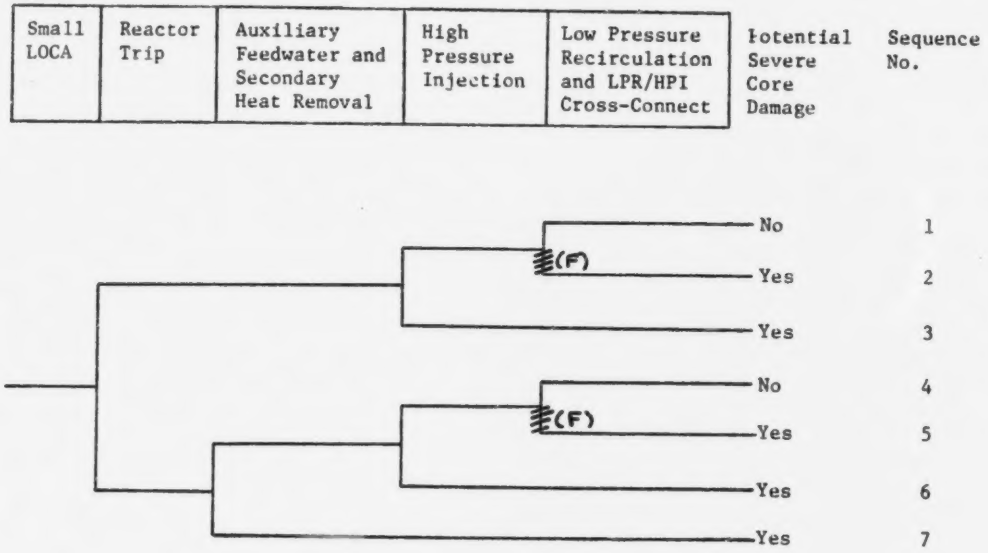
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

BEST DOCUMENT AVAILABLE


Unit in cold shutdown
Purification system & shutdown cooling system cross-connected
Resin transfer using air
Air leak into shutdown cooling system
Both LPSI pumps airbound
Potential Severe Core Damage
Yes
No
No (in cold shutdown)
No
No
No
No

NSIC 141523 — Actual Occurrence for Both LPSI Pumps Airbound at Calvert Cliffs 2


Small LOCA
Reactor Trip
Auxiliary Feedwater and Secondary Heat Removal
High Pressure Injection
Low Pressure Recirculation and LPR/HPI Cross-Connect
Potential Severe Core Damage
Sequence No.
(F)
(F)
No 1
Yes 2
Yes 3
No 4
Yes 5
Yes 6
Yes 7

NSIC 141523 — Sequence of Interest for Both LPSI Pumps Airbound at Calvert Cliffs 2

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 141523

DATE OF LER: October 31, 1978

DATE OF EVENT: October 17, 1978

SYSTEM INVOLVED: LPSI

COMPONENT INVOLVED: Pumps

CAUSE: Pumps airbound. Air leak into shutdown cooling system.

SEQUENCE OF INTEREST: LOCA

ACTUAL OCCURRENCE: Resin transfer using air

REACTOR NAME: Calvert Cliffs 2

DOCKET NUMBER: 50-318

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 845 MWe

REACTOR AGE: 2.2 yr

VENDOR: Combustion Engineering

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Baltimore Gas & Electric Co.

LOCATION: 40 miles south of Annapolis, Md

DURATION: 2 hours

PLANT OPERATING CONDITION: Cold shutdown

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ________________

DISCOVERY METHOD: operational event

COMMENT: Following this type of operation LPSI pumps should be vented to assure operability.

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 141706

Date: October 24, 1978

Title: Diesel Generator Inoperable at Rancho Seco

The failure sequence was:

1. With the plant at power, maintenance was being performed on diesel generator B.
2. While diesel generator A was running, a loss of voltage regulation occurred on the diesel, due to the failure of fuses in the control power circuit for the voltage regulator.

Corrective action:

1. Plant shutdown was commenced.
2. The failed fuses were replaced, and the reactor returned to full power.

Design purpose of failed system or component:

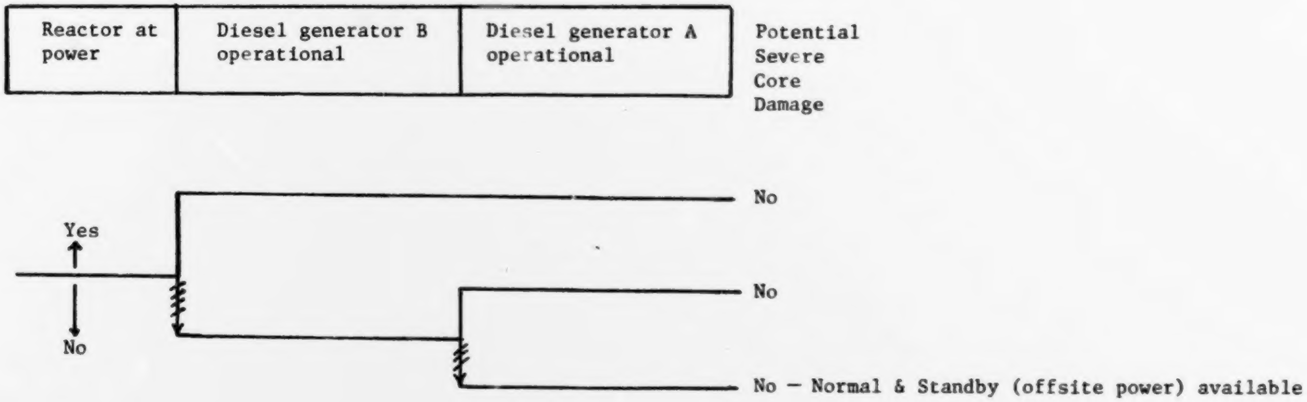
1. The fuses provide overcurrent protection for the diesel generator control power circuits.
2. The diesel generator provides a source of emergency on-site power in the event of loss of off-site power.

Unavailability of system per WASH 1400:* emergency power system: 1 × 10^{-2}/D

Unavailability of component per WASH 1400:* emergency diesel generator: 3 × 10^{-2}/D
fuse, premature open: 1 × 10^{-6}/hr

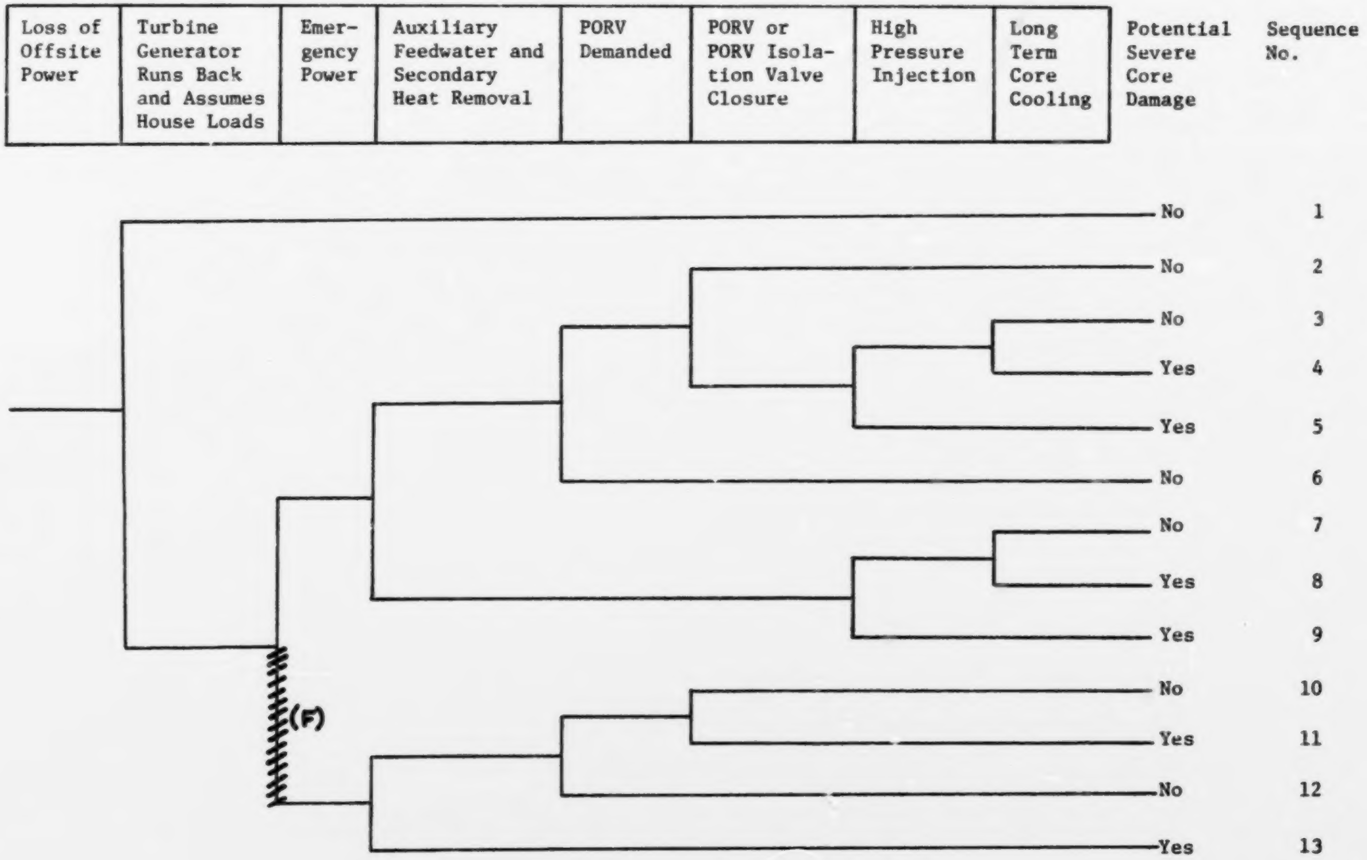
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor at power
Diesel generator B operational
Diesel generator A operational
Potential Severe Core Damage
Yes
No
No
No
No — Normal & Standby (offsite power) available

NSIC 141706 — Actual Occurrence for Loss of Emergency Power at Rancho Seco

533


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
(F)
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
Yes 9
No 10
Yes 11
No 12
Yes 13

NSIC 141706 — Sequence of Interest for Loss of Emergency Power at Rancho Seco

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 141706

DATE OF LER: November 6, 1978

DATE OF EVENT: October 24, 1978

SYSTEM INVOLVED: On-site emergency power

COMPONENT INVOLVED: diesel generator

CAUSE: Loss of voltage regulation due to fuse failure.

SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE: Loss of one diesel generator while the second was out-of-service.

REACTOR NAME: Rancho Seco

DOCKET NUMBER: 50-312

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 918 MWe

REACTOR AGE: 4.2 yr

VENDOR: B&W

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Sacramento Municipal Utility District

LOCATION: 25 miles SE of Sacramento, CA

DURATION: 7 hours

PLANT OPERATING CONDITION: at 100% power.

SAFETY FEATURE TYPE OF FAILURE: [✓ (a)] inadequate performance; (b) failed to start; (c) made inoperable; (d) ________________

DISCOVERY METHOD: Operational event.

COMMENT: –

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 142329

Date: December 8, 1978

Title: Blockage Found in BIT Discharge Line at Indian Point 2

The failure sequence was:

1. A wire in the heat trace circuit for the Boron Injection Tank discharge piping failed.
2. The lack of heat tracing resulted in boric acid solidification, which plugged the discharge line.

Corrective action;

1. The defective wire was replaced.
2. Test procedures were revised to require more frequent tests of the affected heat tracing circuits.

Design purpose of failed system or component:

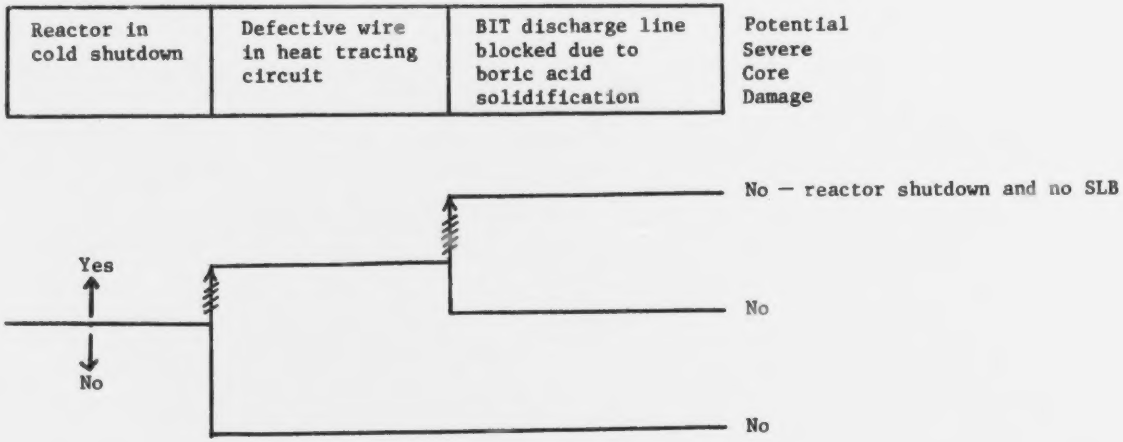
1. The heat tracing circuits keep the temperature of the piping containing concentrated boric acid at a temperature above the boric acid solidification point.
2. The BIT provides a source of concentrated boric acid for reactor shutdown following a steam line break.

Unavailability of system per WASH 1400:* High Pressure Injection System: $1.2 \times 10^{-2}/D$

Unavailability of component per WASH 1400:* heat tracing circuit, open: $4 \times 10^{-6}/D$

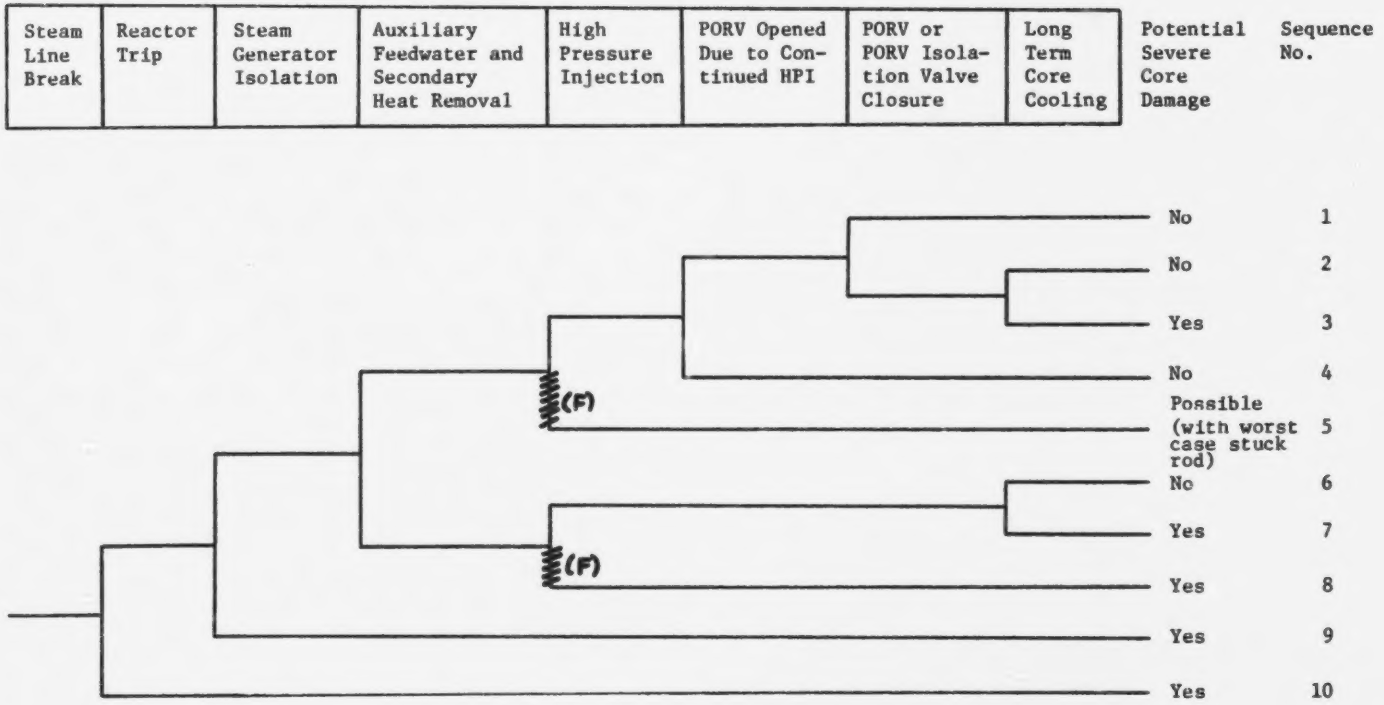
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor in cold shutdown
Defective wire in heat tracing circuit
BIT discharge line blocked due to boric acid solidification
Potential Severe Core Damage
Yes
No
No — reactor shutdown and no SLB
No
No

NSIC 142329 — Actual Occurrence for Blockage Found in BIT Discharge Line at Indian Point 2

BEST DOCUMENT AVAILABLE


Steam Line Break
Reactor Trip
Steam Generator Isolation
Auxiliary Feedwater and Secondary Heat Removal
High Pressure Injection
PORV Opened Due to Continued HPI
PORV or PORV Isolation Valve Closure
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
Yes 3
No 4
(F)
Possible (with worst case stuck rod) 5
No 6
Yes 7
(F)
Yes 8
Yes 9
Yes 10

NSIC 142329 — Sequence of Interest for Blockage Found in BIT Discharge Line at Indian Point 2

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 142329

DATE OF LER: December 8, 1978

DATE OF EVENT: November 26, 1978

SYSTEM INVOLVED: Safety Injection System (High Pressure Injection)

COMPONENT INVOLVED: BIT Discharge Piping

CAUSE: Failure of a heat tracing circuit

SEQUENCE OF INTEREST: Steam Line Break

ACTUAL OCCURRENCE: Heat Tracing Circuit Failure

REACTOR NAME: Indian Point 2

DOCKET NUMBER: 50-247

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 873 MWe

REACTOR AGE: 7. yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: United Engineers & Constructors

OPERATORS: Consolidated Edison Co. of New York

LOCATION: 25 miles north of New York, NY

DURATION: 360(a) hours

PLANT OPERATING CONDITION: Cold shutdown

SAFETY FEATURE TYPE OF FAILURE: [✓ (a)] inadequate performance; (b) failed to start;
(c) made inoperable; (d) ____________

DISCOVERY METHOD: via control room instrumentation

COMMENT: Injection of concentrated boric acid following a steam line break terminates reactor return to power and subsequent core damage.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 142462

Date: December 11, 1978

Title: Loss of a Vital Instrument Bus Initiates a Reactor Trip at Salem 1

The failure sequence was:

1. While at 100% power, vital instrument bus 1B was lost due to the failure of an output transformer and two regulating resistors.
2. Loss of the vital bus caused a false low RCS loop flow signal, which caused a reactor trip.
3. AFW pump 12 failed to start due to the loss of vital bus 1B.
4. AFW pump 13 failed to start due to a misadjustment of the overspeed trip mechanism.
5. Decreasing T_{ave} plus safety injection inputs caused by the loss of the 1B vital bus resulted in an inadvertant safety injection.

(See additional page)

Corrective action;

1. AFW pump 12 was manually started.
2. The vital bus was reenergized from its emergency source.
3. The failed inverter components were replaced.
4. The overspeed trip mechanism was re-adjusted.
5. The failed RHR pump breaker was replaced.

Design purpose of failed system or component:

1. The vital bus provides regulated, uninterruptible power for reactor plant instrumentation.
2. The auxiliary feedwater pumps provide feed water for steam generator/reactor cooling when the main feed system is unavailable.
3. The RHR pumps provide for reactor heat removal during cold shutdown and also function as low pressure injection pumps during LOCA mitigation.

Unavailability of system per WASH 1400:* Insufficient power on vital bus: 1.7×10^{-5}/hr

Unavailability of component per WASH 1400:* pump, failure to start: 10^{-3}/D
breaker failure to operate: 10^{-3}/D
transformers: 1×10^{-6}/hr

---

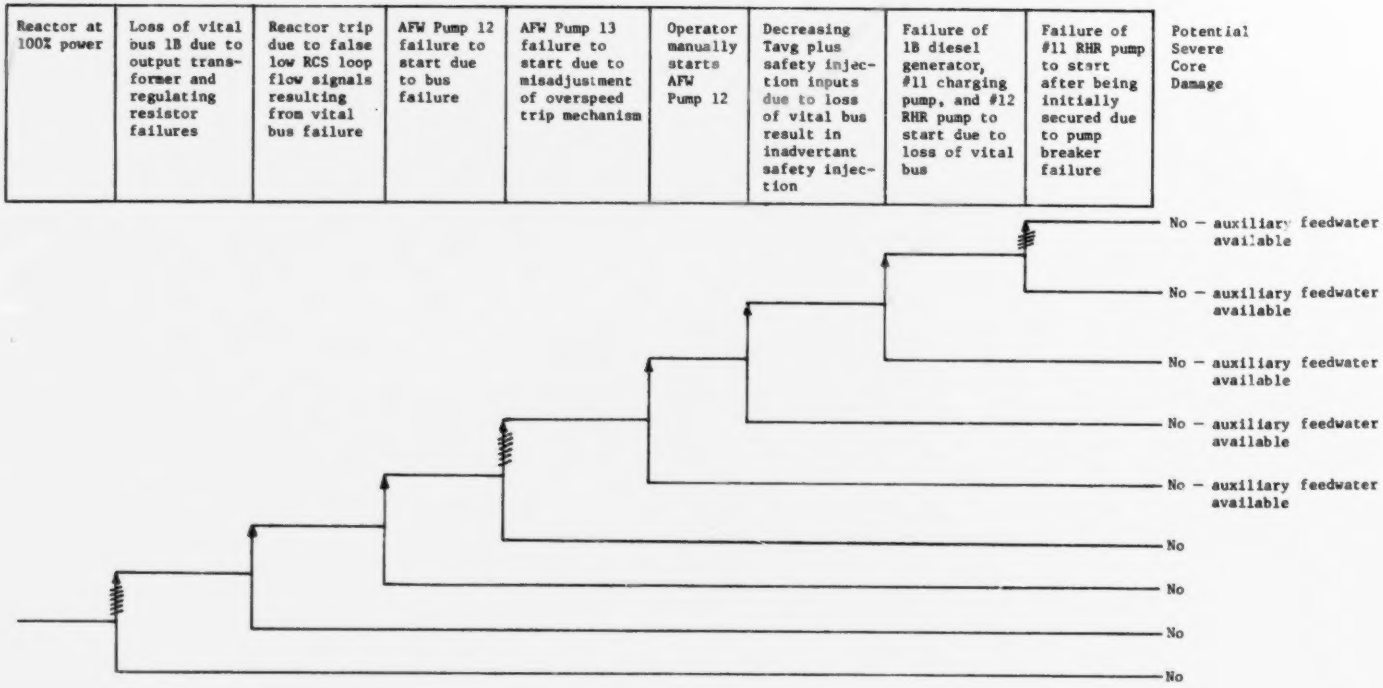
*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

[redacted]

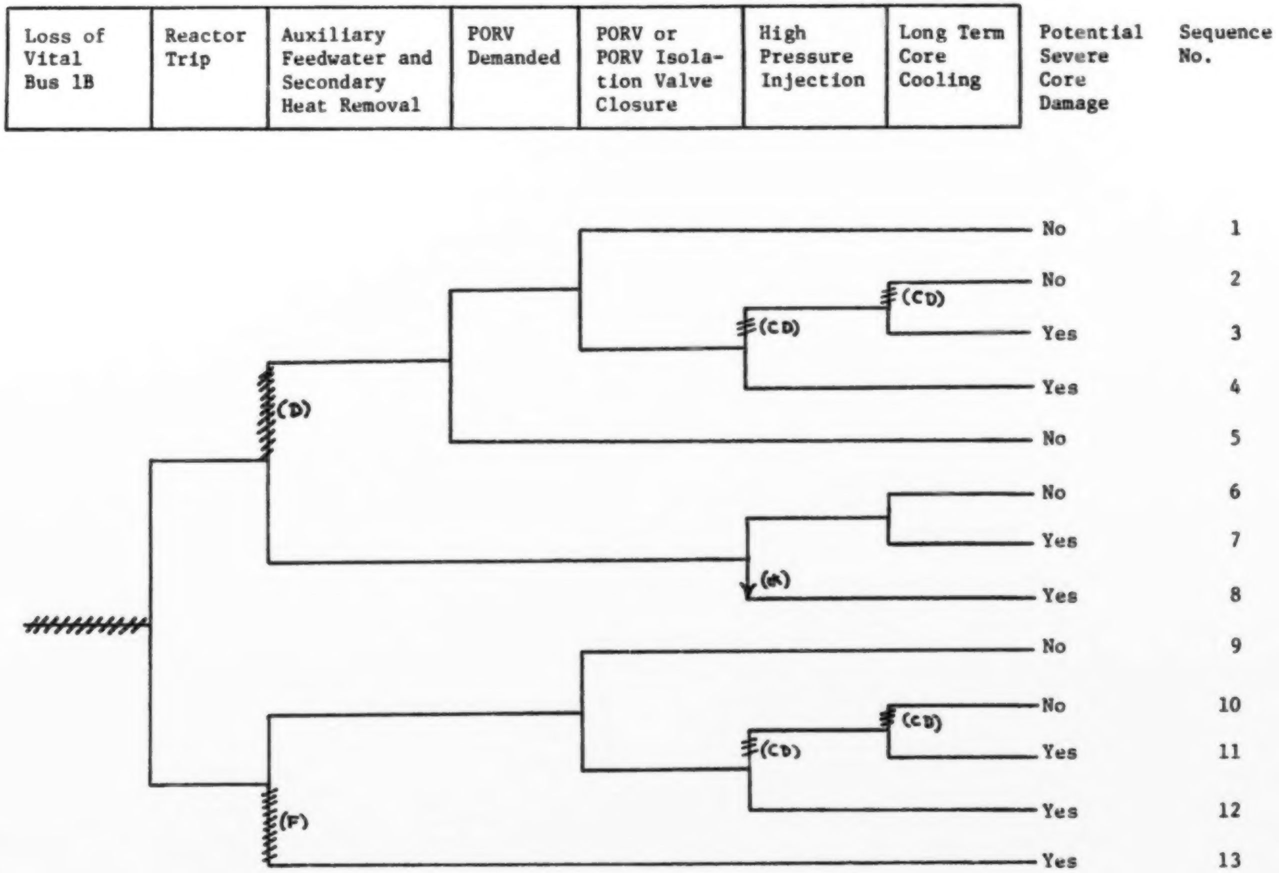
The failure sequence was: (Continued)

6. The 1B diesel generator, #11 charging pump, and #12 RHR pump failed to start due to the loss of the vital bus.
7. After initially securing the #11 RHR pump, it failed to start when a restart was attempted due to a breaker failure.

BEST DOCUMENT AVAILABLE


Reactor at 100% power
Loss of vital bus 1B due to output transformer and regulating resistor failures
Reactor trip due to false low RCS loop flow signals resulting from vital bus failure
AFW Pump 12 failure to start due to bus failure
AFW Pump 13 failure to start due to misadjustment of overspeed trip mechanism
Operator manually starts AFW Pump 12
Decreasing Tavg plus safety injection inputs due to loss of vital bus result in inadvertant safety injection
Failure of 1B diesel generator, #11 charging pump, and #12 RHR pump to start due to loss of vital bus
Failure of #11 RHR pump to start after being initially secured due to pump breaker failure
Potential Severe Core Damage
No – auxiliary feedwater available
No – auxiliary feedwater available
No – auxiliary feedwater available
No – auxiliary feedwater available
No – auxiliary feedwater available
No
No
No
No

NSIC 142462 – Actual Occurrence for Loss of a Vital Instrument Bus Initiates a Reactor Trip at Salem 1


Loss of Vital Bus 1B
Reactor Trip
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
(D)
(CD)
(CD)
(*)
(CD)
(CD)
(F)
No 1
No 2
Yes 3
Yes 4
No 5
No 6
Yes 7
Yes 8
No 9
No 10
Yes 11
Yes 12
Yes 13

NSIC 142462 — Sequence of Interest for Loss of a Vital Instrument Bus Initiates a Reactor Trip at Salem 1

*Not included in mitigation procedures.

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 142462

DATE OF LER: December 11, 1978

DATE OF EVENT: November 27, 1978

SYSTEM INVOLVED: Vital power, auxiliary feedwater, low pressure injection

COMPONENT INVOLVED: Inverter, AFW pump, breaker

CAUSE: The inverter failed due to a failed output transformer and regulating resistors. The AFW pump failed because of a mis-adjusted overspeed trip, and the breaker failed to close.

SEQUENCE OF INTEREST: Loss of vital instrument bus

ACTUAL OCCURRENCE: Loss of vital instrument bus and consequent reactor trip and safety injection.

REACTOR NAME: Salem 1

DOCKET NUMBER: 50-272

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 1090 MWe

REACTOR AGE: 2.3 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Utility

OPERATORS: Public Service Electric & Gas Co.

LOCATION: 20 miles south of Wilmington, DE

DURATION: N/A

PLANT OPERATING CONDITION: 100% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; [✓ (b)] failed to start; [✓ (c)] made inoperable; (d) ____________

DISCOVERY METHOD: operational transient

COMMENT: –

544

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 143219

Date: August 4, 1977

Title: Offsite Power to Station Lost at Indian Point 3

The failure sequence was:

1. While at hot shutdown following a reactor trip, offsite power was lost due to severe electrical storms.
2. The emergency generators started and assumed required safety-related loads.

Corrective action:

1. Offsite power was restored.

Design purpose of failed system or component:

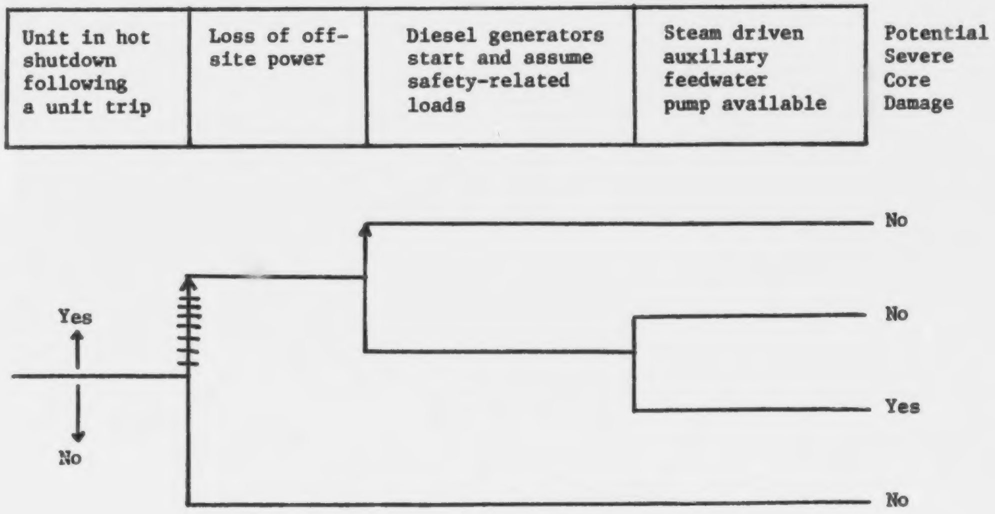
Offsite power provides the preferred source of electric power for safety-related loads on this plant.

Unavailability of system per WASH 1400:* offsite power: 2×10^{-5}/hr

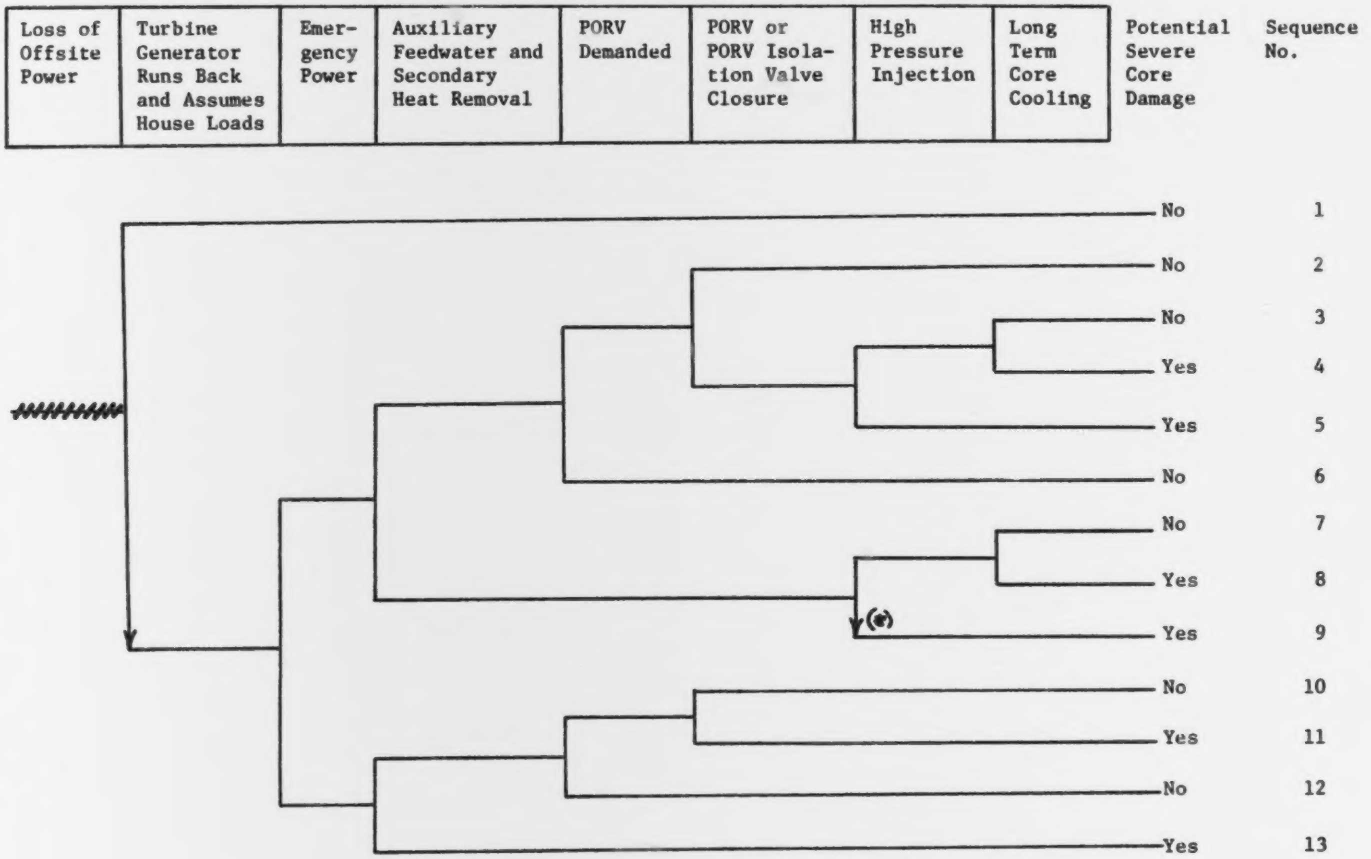
Unavailability of component per WASH 1400:*

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Unit in hot shutdown following a unit trip
Loss of off-site power
Diesel generators start and assume safety-related loads
Steam driven auxiliary feedwater pump available
Potential Severe Core Damage
Yes
No
No
No
Yes
No

NSIC 143219 — Actual Occurrence for Loss of Offsite Power at Indian Point 3


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
(*)
Yes 9
No 10
Yes 11
No 12
Yes 13

NSIC 143219 — Sequence of Interest for Loss of Offsite Power at Indian Point 3

*not included in mitigation procedures.

BEST DOCUMENT AVAILABLE

547

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 143219

DATE OF LER: August 4, 1977

DATE OF EVENT: July 13, 1977

SYSTEM INVOLVED: Offsite power

COMPONENT INVOLVED: Transmission network

CAUSE: Severe electrical storms

SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE: Loss of offsite power

REACTOR NAME: Indian Point 3

DOCKET NUMBER: 50-286

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 965 MWe

REACTOR AGE: 1.8 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: United Engineers & Constructors

OPERATORS: Consolidated Edison Co. of New York

LOCATION: 25 miles north of New York, NY

DURATION: N/A

PLANT OPERATING CONDITION: Hot shutdown

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ________________

DISCOVERY METHOD: During operation

COMMENT: –

548

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 143255

Date: August 29, 1977

Title: Two Diesels Fail to Reach Speed When Starting at Salem 1

The failure sequence was:

1. With the reactor in hot shutdown, diesel-generators 1A and 1B failed to start due to a lack of lubrication on the fuel rack linkages.

Corrective action;

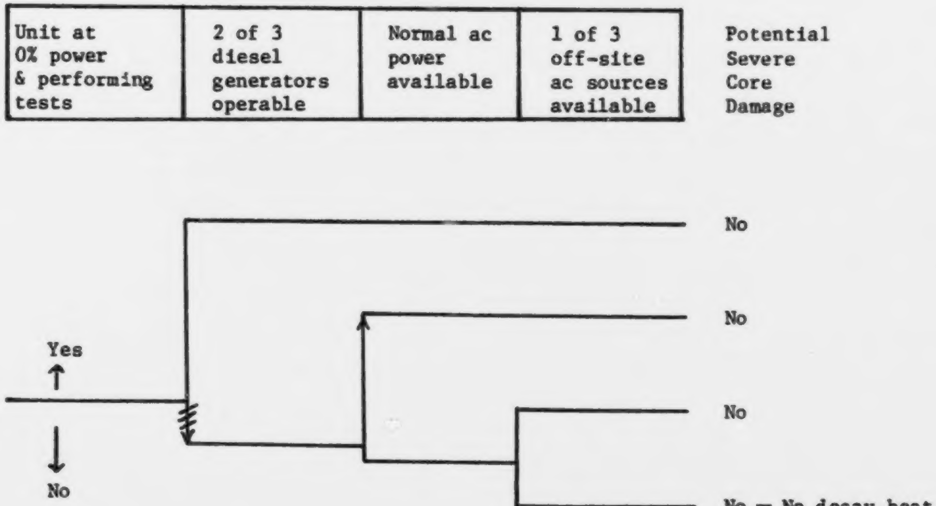
1. The fuel rack linkages were lubricated and the diesels test run satisfactorily.

Design purpose of failed system or component:

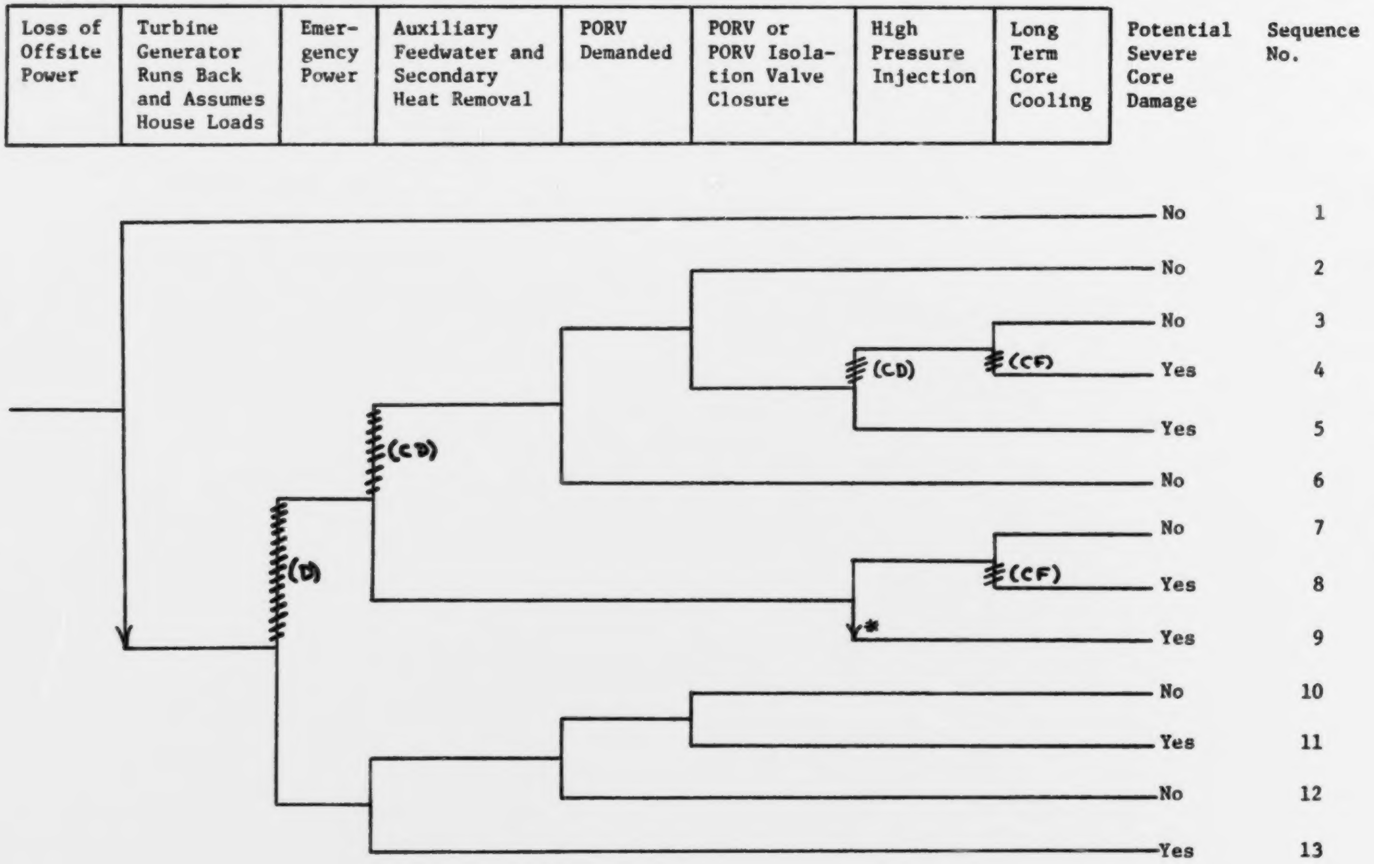
1. The diesel generators provide power to safety-related equipment in the event of loss of offsite power.

Unavailability of system per WASH 1400: Emergency power: 1 × 10^{-2}/D

Unavailability of component per WASH 1400: Diesel generator: 3 × 10^{-2}/D


Unit at 0% power & performing tests
2 of 3 diesel generators operable
Normal ac power available
1 of 3 off-site ac sources available
Potential Severe Core Damage
No
No
No
No — No decay heat
Yes
No

NSIC 143255 — Actual Occurrence for 2 of 3 Diesel Generators Failing to Reach Rated Speed at Salem 1


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
(D)
(CD)
(CD)
(CF)
(CF)
*
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
Yes 9
No 10
Yes 11
No 12
Yes 13

NSIC 143255 – Sequence of Interest for 2 of 3 Diesel Generators Fail to Reach Rated Speed at Salem 1

* not included in mitigation procedures.

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 143255

DATE OF LER: August 29, 1977

DATE OF EVENT: July 30, 1977

SYSTEM INVOLVED: Emergency power

COMPONENT INVOLVED: Diesel-generators

CAUSE: Lack of lubrication of the fuel rack linkages

Sequence of Interest:	Reactor trip with loss of offsite power
Actual Occurrence:	Diesel-generators failure to start during testing.

REACTOR NAME: Salem 1

DOCKET NUMBER: 50-272

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 1090 MWe

REACTOR AGE: 1.0 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Utility

OPERATORS: Public Service Electric & Gas Co.

LOCATION: 20 miles south of Wilmington, DE

DURATION: 4.75 hours

PLANT OPERATING CONDITION: hot shutdown

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; [✓ (b)] failed to start; (c) made inoperable; (d) ________________

DISCOVERY METHOD: during testing

COMMENT: –

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: **143296**

Date: **October 20, 1977**

Title: **Both RWST Outlet Valves Inadvertantly Closed at Maine Yankee**

The failure sequence was:

1. **During surveillance testing of ECCS pumps and valves, an operator error resulted in the closure of both RWST valves to the LPSI and containment spray pumps.**
2. **During subsequent LPSI operability checks, the pumps failed to develop adequate discharge head.**

Corrective action:

The valves were reopened once they were discovered closed.

Design purpose of failed system or component:

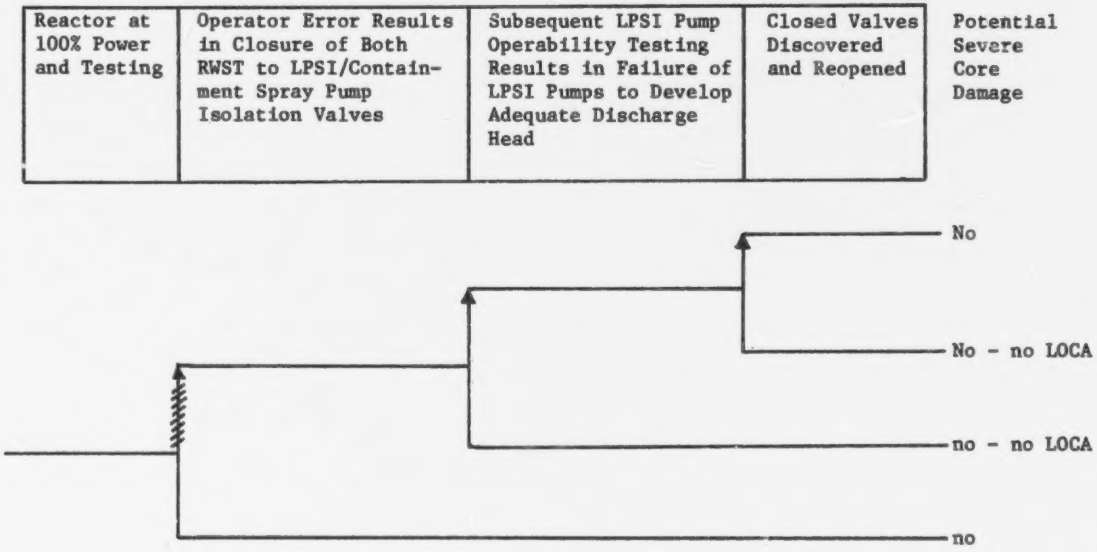
The valves isolate the RWST from the LPSI and containment spray pump suction during recirculation cooling following a LOCA and during maintenance.

Unavailability of system per WASH 1400:* **Low pressure injection: 4.7×10^{-3}/D**

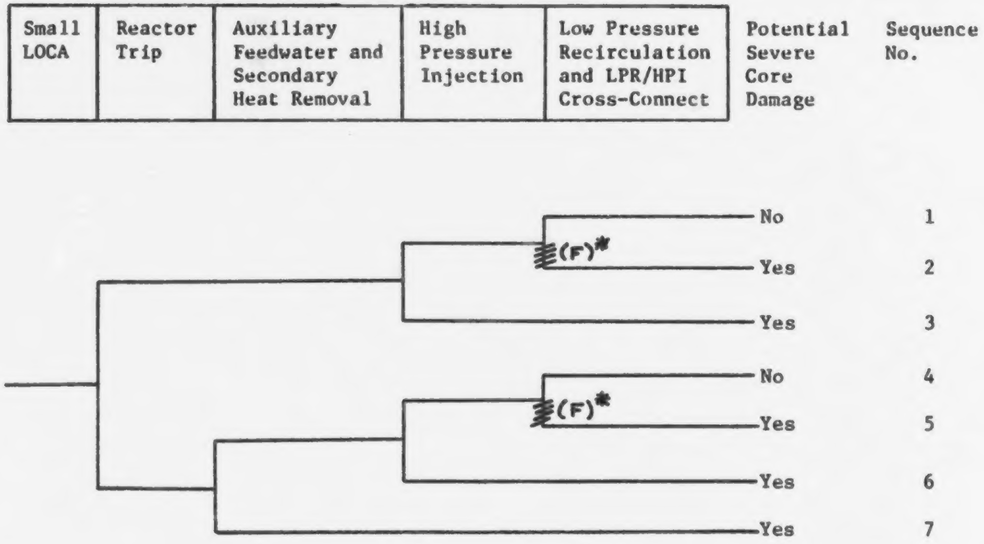
Unavailability of component per WASH 1400:* –

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor at 100% Power and Testing
Operator Error Results in Closure of Both RWST to LPSI/Containment Spray Pump Isolation Valves
Subsequent LPSI Pump Operability Testing Results in Failure of LPSI Pumps to Develop Adequate Discharge Head
Closed Valves Discovered and Reopened
Potential Severe Core Damage
No
No - no LOCA
no - no LOCA
no

NSIC 143296 — Actual Occurrence for Both RWST Outlet Valves Inadvertantly Closed at Maine Yankee


Small LOCA
Reactor Trip
Auxiliary Feedwater and Secondary Heat Removal
High Pressure Injection
Low Pressure Recirculation and LPR/HPI Cross-Connect
Potential Severe Core Damage
Sequence No.
(F)*
No
1
Yes
2
Yes
3
No
4
(F)*
Yes
5
Yes
6
Yes
7

NSIC 143296 — Sequence of Interest for Both RWST Outlet Valves Inadvertantly Closed at Maine Yankee

*success requires operator detection of closed valves and their re-opening prior to LPI pump failure from cavitation.

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 143296

DATE OF LER: October 27, 1977

DATE OF EVENT: October 20, 1977

SYSTEM INVOLVED: low pressure safety injection

COMPONENT INVOLVED: RWST outlet isolation valves

CAUSE: Valves inadvertently closed during testing, human error

SEQUENCE OF INTEREST: small LOCA

ACTUAL OCCURRENCE: Inadvertent valve closure during testing

REACTOR NAME: Maine Yankee

DOCKET NUMBER: 50-309

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 825 MWe

REACTOR AGE: 5.0 yr

VENDOR: CE

ARCHITECT-ENGINEERS: Stone & Webster

OPERATORS: Maine Yankee Atomic Power Co.

LOCATION: 10 miles north of Bath, Maine

DURATION: .17 hours

PLANT OPERATING CONDITION: 100% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ______________________

DISCOVERY METHOD: During testing

COMMENT: These motor operated valves may receive "open" signals during LPSI system actuation. However, this is not addressed in the FSAR.

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 143533

Date: October 11, 1977

Title: Diesel Generator Load Cycles at Brunswick 1

The failure sequence was:

1. The #4 diesel was under a 7 day L.C.O. for its annual inspection.
2. The Diesel generator #2 was unable to maintain a steady load without cycling about 500 kw during a periodic performance test.
3. A pair of electrical leads in a connector on the woodward governor unit were broken. This also caused an amplifier to short out. The brake was attributed to an incorrect connector which had been installed during construction.

Corrective action:

The amplifier was replaced and the broken wire and connector were repaired.

Design purpose of failed system or component:

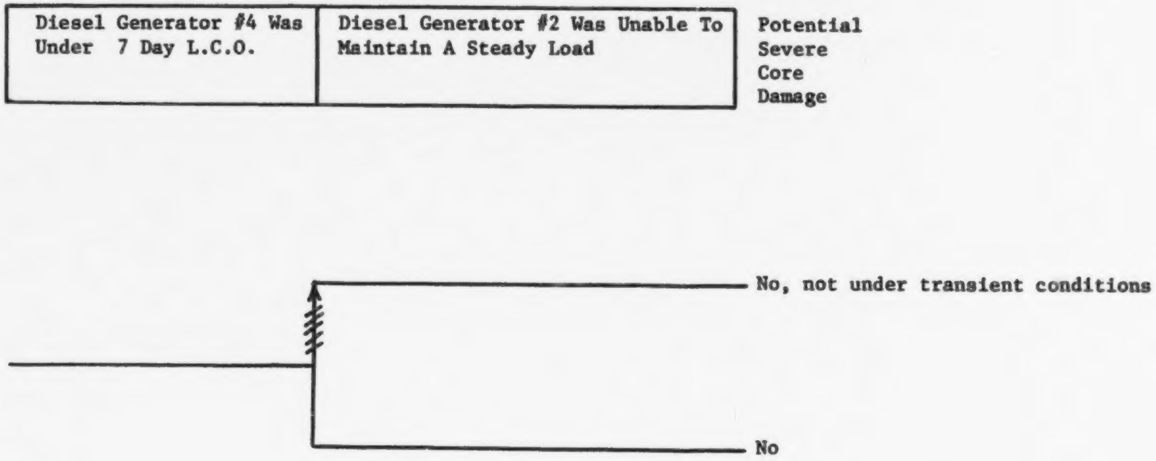
The diesel generators provide a.c. power when offsite sources are unavailable.

Unavailability of system per WASH 1400:* Emergency Power: 10^{-3}/hr

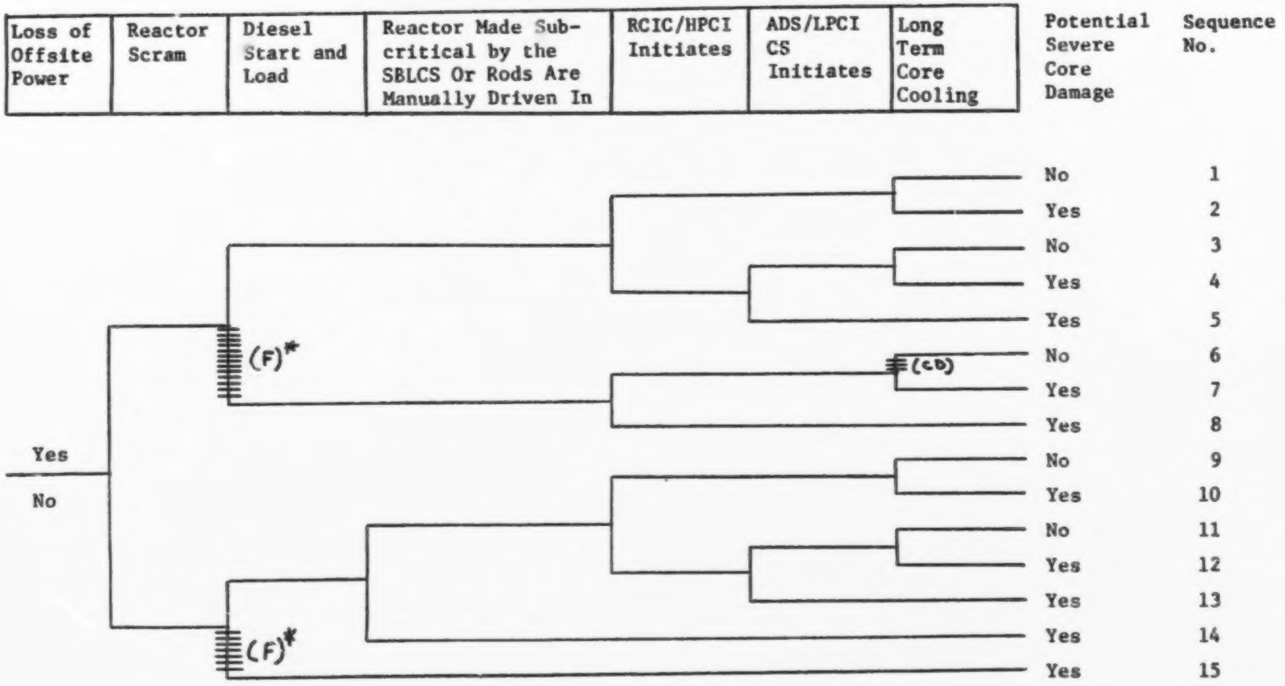
Unavailability of component per WASH 1400:* Wires, Open: 3×10^{-6}/hr
Diesel engine failure to run:
3×10^{-4}/hr

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Diesel Generator #4 Was Under 7 Day L.C.O.
Diesel Generator #2 Was Unable To Maintain A Steady Load
Potential Severe Core Damage
No, not under transient conditions
No

NSIC 143533 — Actual Occurrence of Diesel Generator Load Cycles at Brunswick 1


Loss of Offsite Power
Reactor Scram
Diesel Start and Load
Reactor Made Sub-critical by the SBLCS Or Rods Are Manually Driven In
RCIC/HPCI Initiates
ADS/LPCI CS Initiates
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
(F)*
(CD)
(F)*
No 1
Yes 2
No 3
Yes 4
Yes 5
No 6
Yes 7
Yes 8
No 9
Yes 10
No 11
Yes 12
Yes 13
Yes 14
Yes 15

NSIC 143533 — Sequence of Interest for Diesel Generator Load Cycles at Brunswick 1

*A degraded state of emergency power may be obtained by proper load shedding.

559 **BEST DOCUMENT AVAILABLE**

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 143533

DATE OF LER: November 8, 1977

DATE OF EVENT: October 10, 1977

SYSTEM INVOLVED: Electric Power

COMPONENT INVOLVED: Diesel Generator Governor

CAUSE: Construction fault, mechanical failure

SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE: Diesel Generator Load Cycles

REACTOR NAME: Brunswick 1

DOCKET NUMBER: 50-325

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 821 MWe

REACTOR AGE: 1.0 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: United Engineers

OPERATORS: Carolina Power and Light

LOCATION: Three miles N of Southport

DURATION: 12 (a) hours

PLANT OPERATING CONDITION: 98% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; (c) made inoperable; [✓ (d) failed to run properly]

DISCOVERY METHOD: testing

COMMENT:

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 144187

Date: December 10, 1977

Title: Diesel Generator Found Inoperable at Millstone 1

The failure sequence was:

1. During testing of the emergency diesel with the gas turbine inoperable the diesel was found to be inoperable. No reason was given for its inoperability.

Corrective action:

1. An orderly plant shutdown was commenced.

Design purpose of failed system or component:

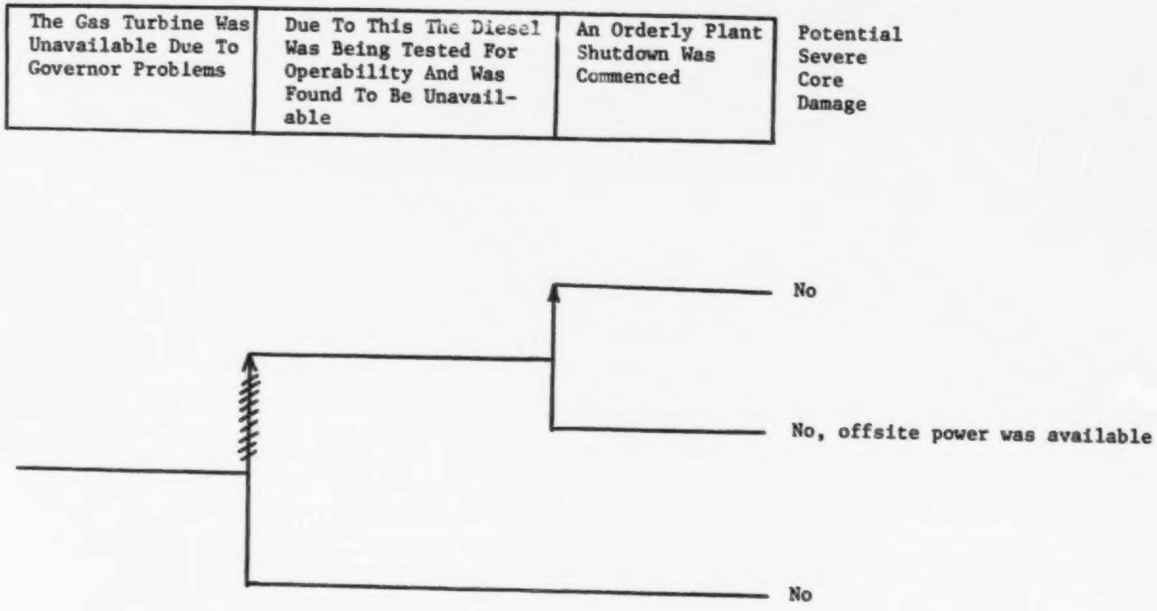
The emergency power system provides a.c. power when offsite sources are unavailable.

Unavailability of system per WASH 1400:* Emergency Power: $1 \times 10^{-3}/D$

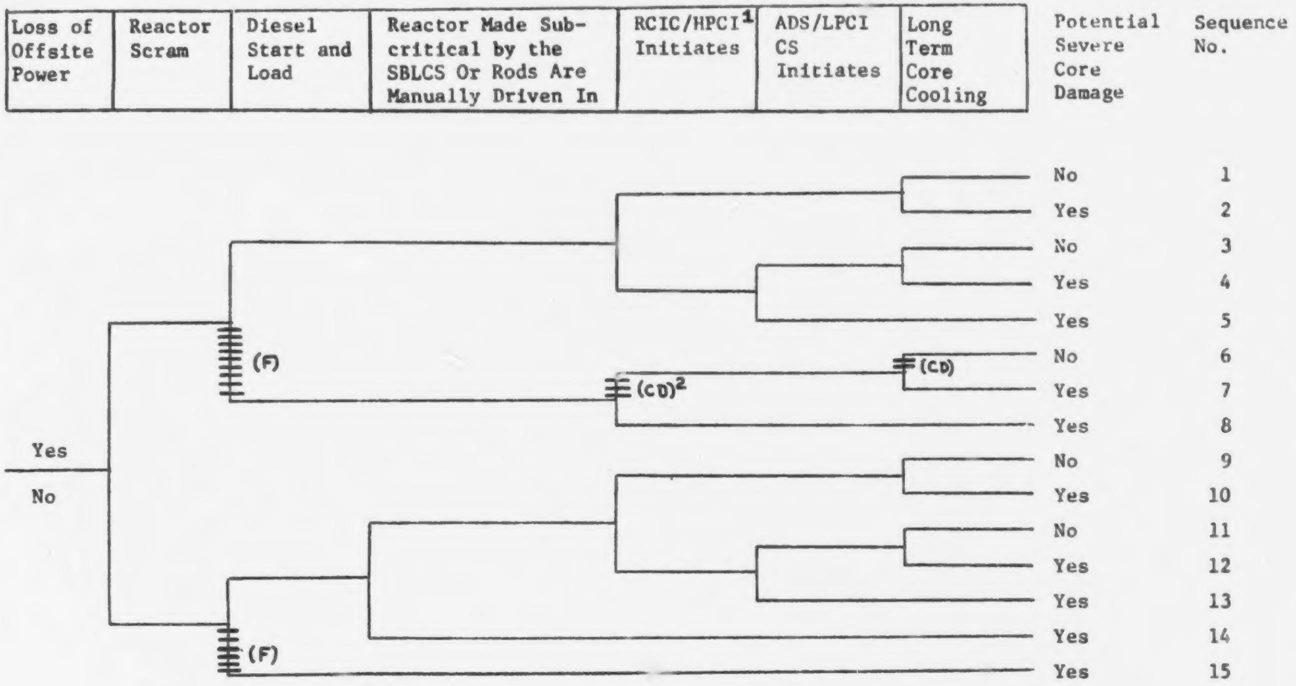
Unavailability of component per WASH 1400:* Diesel Generator failure to start: $3.0 \times 10^{-2}/D$

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


The Gas Turbine Was Unavailable Due To Governor Problems
Due To This The Diesel Was Being Tested For Operability And Was Found To Be Unavailable
An Orderly Plant Shutdown Was Commenced
Potential Severe Core Damage
No
No, offsite power was available
No

NSIC 144187 — Actual Occurrence of Diesel Generator Found Inoperable at Millstone 1


Loss of Offsite Power
Reactor Scram
Diesel Start and Load
Reactor Made Subcritical by the SBLCS Or Rods Are Manually Driven In
RCIC/HPCI[1] Initiates
ADS/LPCI CS Initiates
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
(F)
(CD)[2]
(CD)
(F)
No 1
Yes 2
No 3
Yes 4
Yes 5
No 6
Yes 7
Yes 8
No 9
Yes 10
No 11
Yes 12
Yes 13
Yes 14
Yes 15

NSIC 144187 — Sequence of Interest for Diesel Generator Found Inoperable at Millstone 1

[1] Millstone 1 utilizes FWCI and IC instead of HPCI

[2] Success of the FWCI system requires success of gas turbine and success of the IC

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 144187

DATE OF LER: December 12, 1977

DATE OF EVENT: December 10, 1977

SYSTEM INVOLVED: Emergency Power

COMPONENT INVOLVED: Emergency Diesel, Emergency Gas Turbine

CAUSE: not given

SEQUENCE OF INTEREST: Loop

ACTUAL OCCURRENCE: Diesel Generator found inoperable

REACTOR NAME: Millstone 1

DOCKET NUMBER: 50-245

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 660 MWe

REACTOR AGE: 7.1 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: Ebasco

OPERATORS: North Eastern States Power

LOCATION: Five miles SW of New London, Conn.

DURATION: 12 (a) hours

PLANT OPERATING CONDITION: 98% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________

DISCOVERY METHOD: surveillance testing

COMMENT:

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 145209

Date: December 20, 1978

Title: **Both Power Operated Relief Valves (PORVs) Open During Troubleshooting at Fort Calhoun 1**

The failure sequence was:

1. During a heatup from cold shutdown, one of the two PORV channels failed to respond to system pressure changes.
2. A technician, while troubleshooting the problem, pulled recorder fuses.
3. This caused both PORVs to open.
4. The operator closed both PORV isolation valves.

Corrective action:

1. Since the PORVs were being used for NDT overpressure protection, additional administrative controls were established to ensure an overpressure condition could not occur.
2. System modifications to prevent reoccurrence were being considered.

Design purpose of failed system or component:

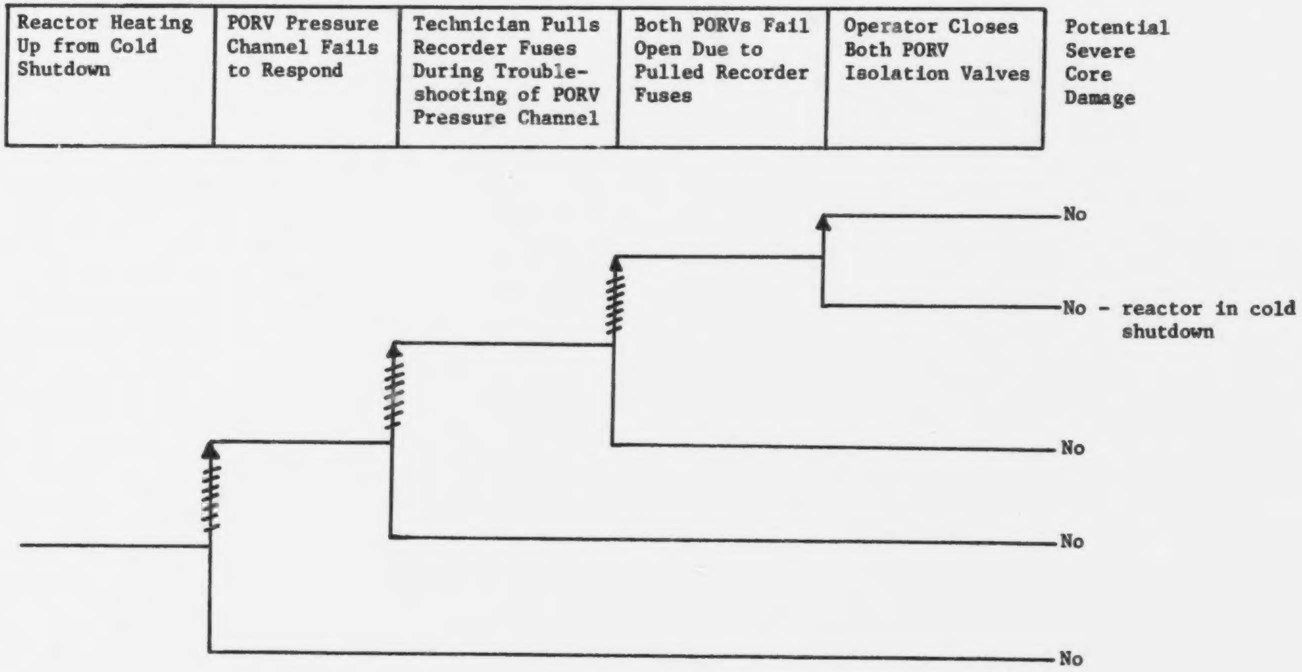
1. During normal operation, the PORVs provide relief protection for the RCS and prevent lifting the pressurizer safety valves during pressure transients.
2. During cold shutdown, the PORVs provide NDT overpressure protection.

Unavailability of system per WASH 1400:* —

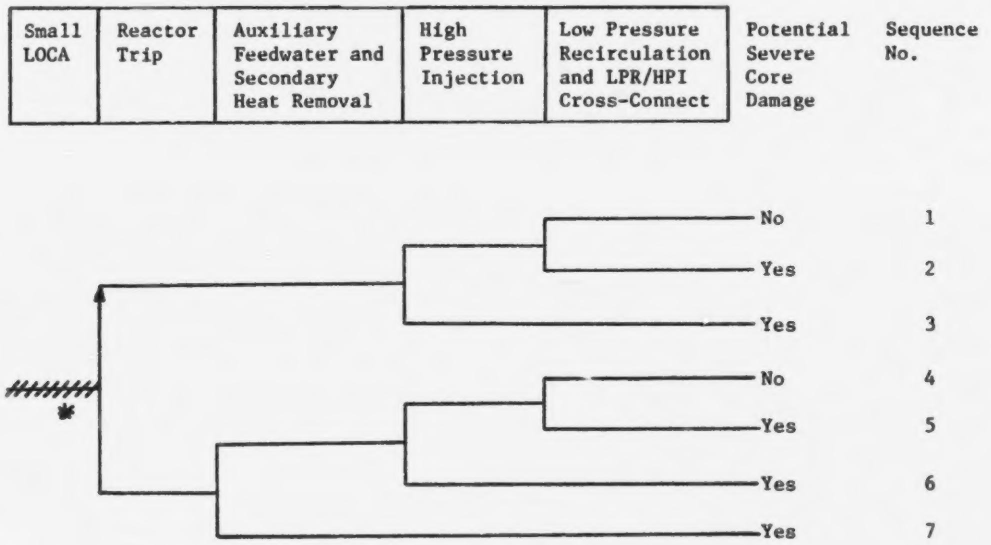
Unavailability of component per WASH 1400:* relief valve, inadvertant opening: 1×10^{-5}/hr

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor Heating Up from Cold Shutdown
PORV Pressure Channel Fails to Respond
Technician Pulls Recorder Fuses During Trouble-shooting of PORV Pressure Channel
Both PORVs Fail Open Due to Pulled Recorder Fuses
Operator Closes Both PORV Isolation Valves
Potential Severe Core Damage
No
No - reactor in cold shutdown
No
No
No

NSIC 145209 — Actual Occurrence for Both Power Operated Relief Valves Open During Troubleshooting at Fort Calhoun 1


Small LOCA
Reactor Trip
Auxiliary Feedwater and Secondary Heat Removal
High Pressure Injection
Low Pressure Recirculation and LPR/HPI Cross-Connect
Potential Severe Core Damage
Sequence No.
*
No
1
Yes
2
Yes
3
No
4
Yes
5
Yes
6
Yes
7

NSIC 145209 — Sequence of Interest for Both Power Operated Relief Valves Open During Troubleshooting at Fort Calhoun 1

[*] initiating event requires operator failure to close PORV isolation valves.

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 145209

DATE OF LER: January 2, 1979

DATE OF EVENT: December 20, 1978

SYSTEM INVOLVED: reactor coolant

COMPONENT INVOLVED: power operated relief valves

CAUSE: Technician error in pulling recorder fuses during troubleshooting resulted in both PORVs failing open, human error

SEQUENCE OF INTEREST: two stuck open PORVs

ACTUAL OCCURRENCE: two stuck open PORVs during cold shutdown

REACTOR NAME: Fort Calhoun 1

DOCKET NUMBER: 50-285

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 457 MWe

REACTOR AGE: 5.5 yr

VENDOR: CE

ARCHITECT-ENGINEERS: Gibbs & Hill, Inc.

OPERATORS: Omaha Public Power District

LOCATION: 19 miles north of Omaha, Nebraska

DURATION: N/A

PLANT OPERATING CONDITION: Cold shutdown

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; (c) made inoperable; [✓ (d)] failed open

DISCOVERY METHOD: During operation

COMMENT: —

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 146540

Date: January 10, 1979

Title: Both Diesel Generators Out of Service at Kewaunee

The failure sequence was:

1. With the reactor at 100% power, diesel generator 1A was started to verify operability prior to placing diesel generator 1B out of service for maintenance.
2. Diesel generator 1B was removed from service for maintenance.
3. Eighty minutes after diesel generator 1B was removed from service, diesel generator 1A failed to run due to a loss of control power resulting from a blown fuse.

Corrective action:

1. Diesel generator 1A's control circuitry was inspected and no problem could be found. The blown fuse was replaced and the diesel generator was tested and returned to service.

Design purpose of failed system or component:

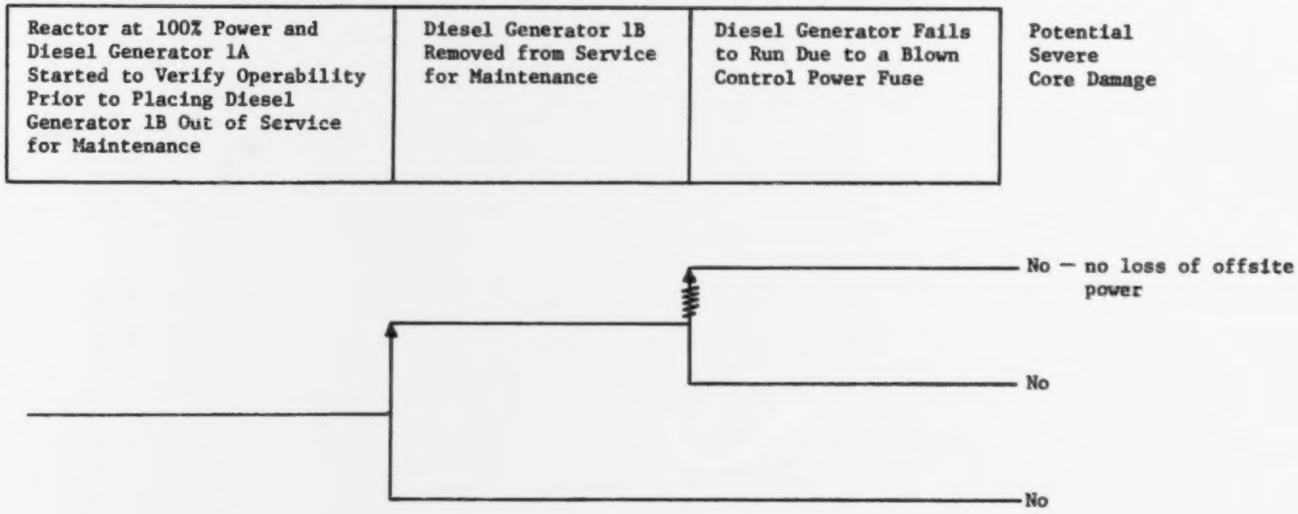
The diesel generators provide power to safety-related loads when the offsite sources and the unit generator are unavailable.

Unavailability of system per WASH 1400:* emergency power: 1×10^{-2}/D

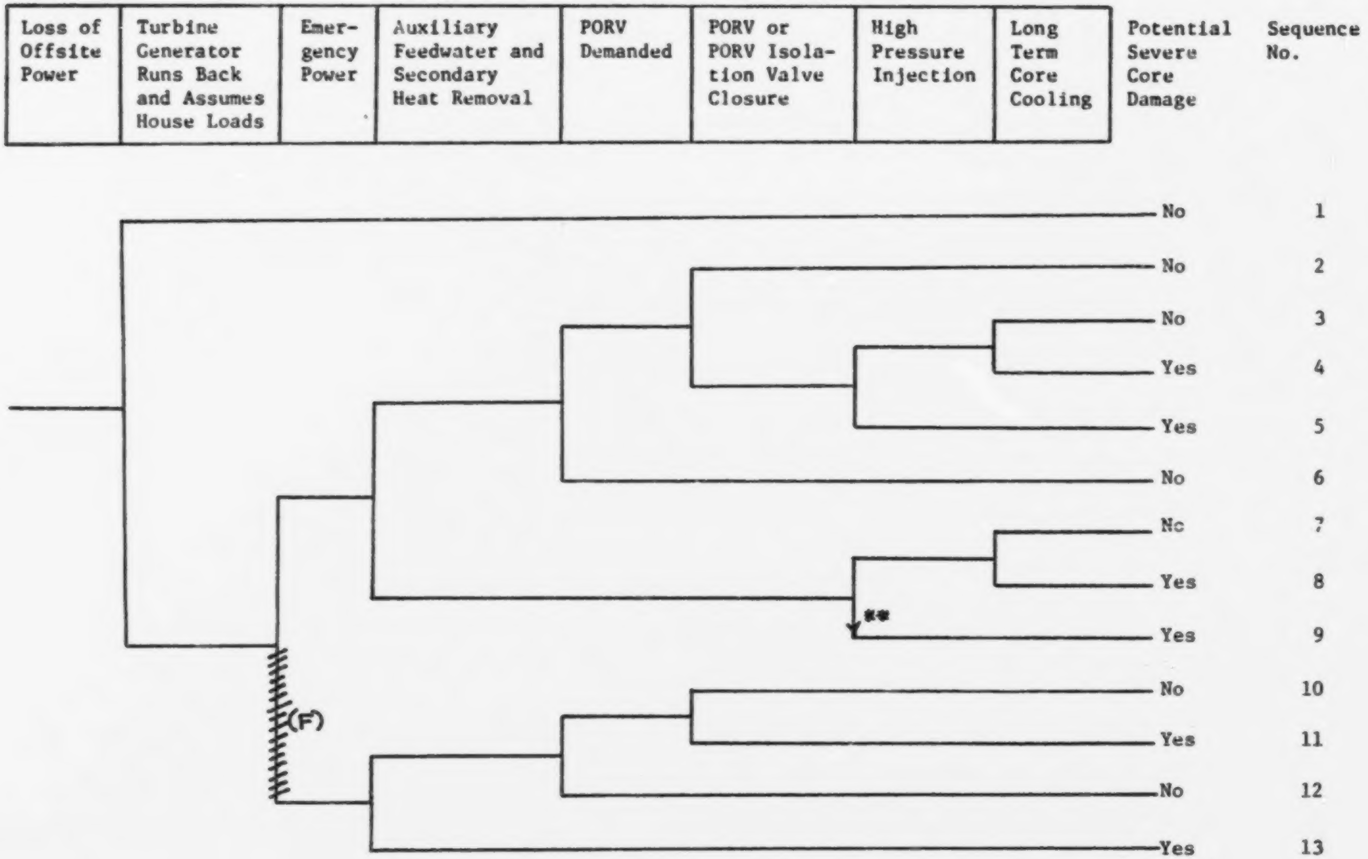
Unavailability of component per WASH 1400:* diesel generator: 3×10^{-2}/D

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor at 100% Power and Diesel Generator 1A Started to Verify Operability Prior to Placing Diesel Generator 1B Out of Service for Maintenance
Diesel Generator 1B Removed from Service for Maintenance
Diesel Generator Fails to Run Due to a Blown Control Power Fuse
Potential Severe Core Damage
No — no loss of offsite power
No
No

NSIC 146540 — Actual Occurrence for Both Diesel Generators Out of Service at Kewaunee


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
Yes 9
No 10
Yes 11
No 12
Yes 13
(F)
**

NSIC 146540 — Sequence of Interest for Both Diesel Generators Out of Service at Kewaunee

*sucess requires manual opening of valves associated with turbine driven AFW pump.

**not included in mitigation procedures.

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 146540

DATE OF LER: January 24, 1979

DATE OF EVENT: January 10, 1979

SYSTEM INVOLVED: emergency power

COMPONENT INVOLVED: diesel generator

CAUSE: blown control power fuse

SEQUENCE OF INTEREST: loss of offsite power

ACTUAL OCCURRENCE: diesel generator failure to run while the other diesel generator was unavailable due to maintenance.

REACTOR NAME: Kewaunee

DOCKET NUMBER: 50-305

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 535 MWe

REACTOR AGE: 5.8 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Flour Power Services Corp.

OPERATORS: Wisconsin Public Service Corp.

LOCATION: 27 miles east of Green Bay, Wisconsin

DURATION: .5 hours

PLANT OPERATING CONDITION: 100% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; (c) made inoperable; [✓ (d)] failed to run

DISCOVERY METHOD: control room annunciation

COMMENT: —

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 146744

Date: January 12, 1979

Title: Power Loss to Vital Bus Results in Reactor Scram at Davis-Besse 1

The failure sequence was:

1. With the reactor at power and RPS channel 3 and AFW pump 1-2 in test, an accidental grounding in a hydrogen analyzer resulted in the loss of vital bus Y2.
2. This, in turn, caused a loss of power to RPS channel 2 and SFRCS channel 2, low indicated RCS flow and 50% indicated reactor power.
3. The low RCS flow and 50% indicated power caused the ICS to increase reactor power and decrease feedwater flow. The loss of SFRCS channel 2 caused the S.G. #2 startup control valve to close, resulting in a decreasing level in S.G. #2.
4. The reactor tripped on flux/flow and the SFRCS tripped on low level in S.G. #2.

(See attached page)

Corrective action:

Blown fuses in the Y2 inverter were replaced and the bus returned to service.

Design purpose of failed system or component:

The Y2 bus is one of four 120VAC uninterruptible buses providing power to safety related instrumentation and controls.

Unavailability of system per WASH 1400:* Unavailability of Power on Vital Bus: 2×10^{-5}/hr

Unavailability of component per WASH 1400:* Inverter (High Power Solid State Device): 3×10^{-6}/hr

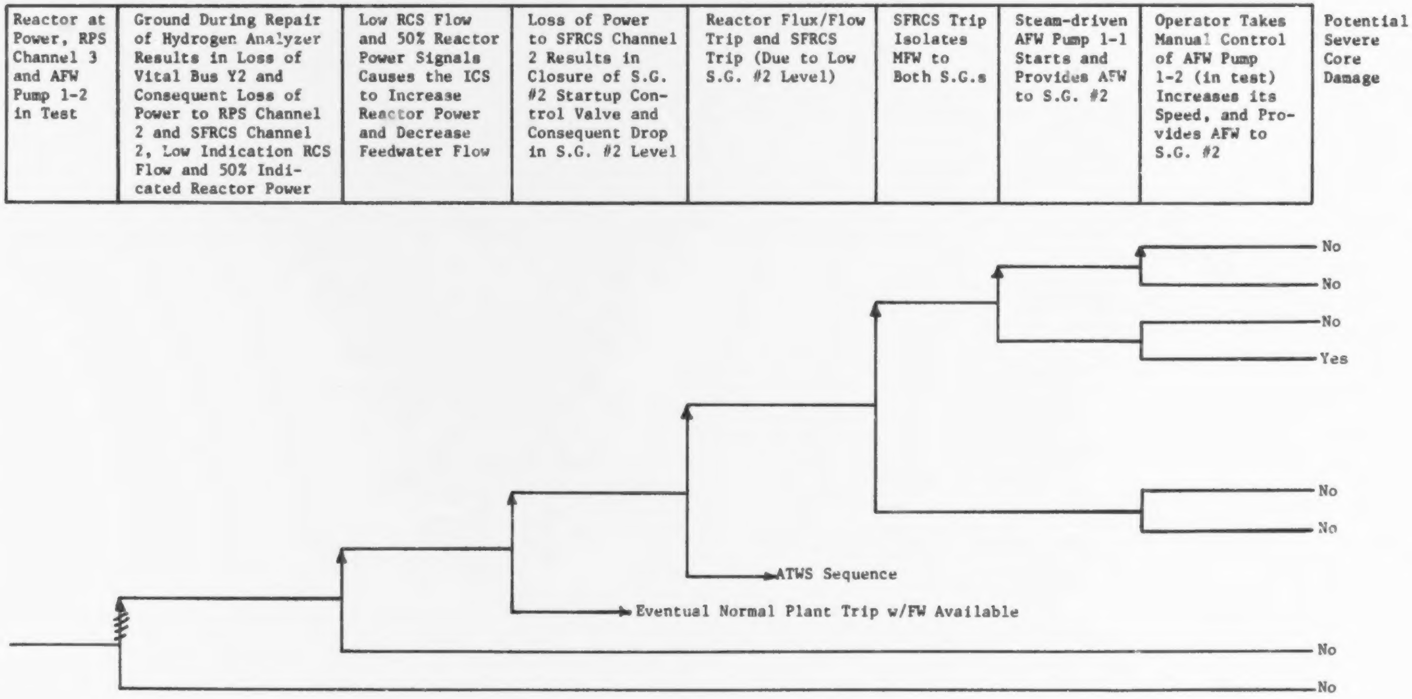
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

[redacted]

The failure sequence was: (continued)

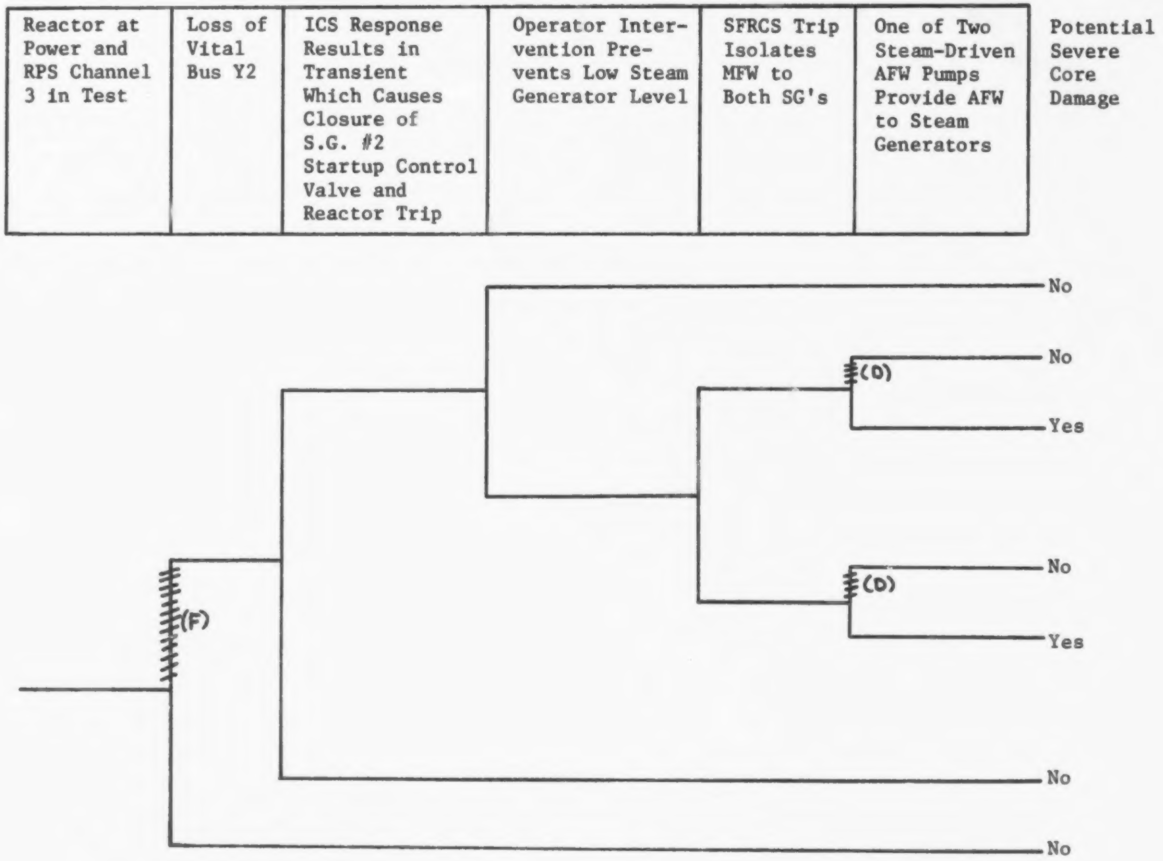
5. The SFRCS trip isolated main feedwater from both steam generators and started auxiliary feedwater pump 1-1, feeding steam generator #1.
6. The operator took manual control of AFW pump 1-2, which was in test, and increased its speed and recovered level in S.G. #2.


Reactor at Power, RPS Channel 3 and AFW Pump 1-2 in Test
Ground During Repair of Hydrogen Analyzer Results in Loss of Vital Bus Y2 and Consequent Loss of Power to RPS Channel 2 and SFRCS Channel 2, Low Indication RCS Flow and 50% Indicated Reactor Power
Low RCS Flow and 50% Reactor Power Signals Causes the ICS to Increase Reactor Power and Decrease Feedwater Flow
Loss of Power to SFRCS Channel 2 Results in Closure of S.G. #2 Startup Control Valve and Consequent Drop in S.G. #2 Level
Reactor Flux/Flow Trip and SFRCS Trip (Due to Low S.G. #2 Level)
SFRCS Trip Isolates MFW to Both S.G.s
Steam-driven AFW Pump 1-1 Starts and Provides AFW to S.G. #2
Operator Takes Manual Control of AFW Pump 1-2 (in test) Increases its Speed, and Provides AFW to S.G. #2
Potential Severe Core Damage
No
No
No
Yes
No
No
ATWS Sequence
Eventual Normal Plant Trip w/FW Available
No
No

NSIC 146744 — Actual Occurrence for Power Loss to Vital Bus Results in Reactor Scram at Davis-Besse

575

BEST DOCUMENT AVAILABLE


Reactor at Power and RPS Channel 3 in Test
Loss of Vital Bus Y2
ICS Response Results in Transient Which Causes Closure of S.G. #2 Startup Control Valve and Reactor Trip
Operator Intervention Prevents Low Steam Generator Level
SFRCS Trip Isolates MFW to Both SG's
One of Two Steam-Driven AFW Pumps Provide AFW to Steam Generators
Potential Severe Core Damage
(F)
(D)
(D)
No
No
Yes
No
Yes
No
No

NSIC 146744 — Sequence of Interest for Power Loss to Vital Bus Results in Reactor Scram at Davis-Besse

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 146744

DATE OF LER: February 8, 1979

DATE OF EVENT: January 12, 1979

SYSTEM INVOLVED: Electric Power (Vital Bus) plus auxiliary feedwater system

COMPONENT INVOLVED: Inverter, Auxiliary Feedwater Pump

CAUSE: Blown fuse in inverter, AFW pump in test, human error

SEQUENCE OF INTEREST: Loss of Vital Bus While at Power

ACTUAL OCCURRENCE: Loss of Vital Bus While at Power

REACTOR NAME: Davis-Besse 1

DOCKET NUMBER: 50-346

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 906 MWe

REACTOR AGE: 1.4 yr

VENDOR: B&W

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Toledo Edison Co.

LOCATION: 21 miles east of Toledo, Ohio

DURATION: N/A

PLANT OPERATING CONDITION: At full power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ______________________

DISCOVERY METHOD: Operational event

COMMENT: –

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 147223

Date: January 26, 1979

Title: Feedwater Block Valve Fails to Close at Crystal River 3

The failure sequence was:

1. During a runback following a turbine trip, a feedwater block valve failed to close when required.
2. The operator manually isolated the feedwater line and tripped the reactor.
3. Reactor cooling was provided via the emergency feedwater system.

Corrective action:

1. The feedwater block valve was found to be stuck on its backseat. The valve was manually operated and then cycled and tested and returned to service.

Design purpose of failed system or component:

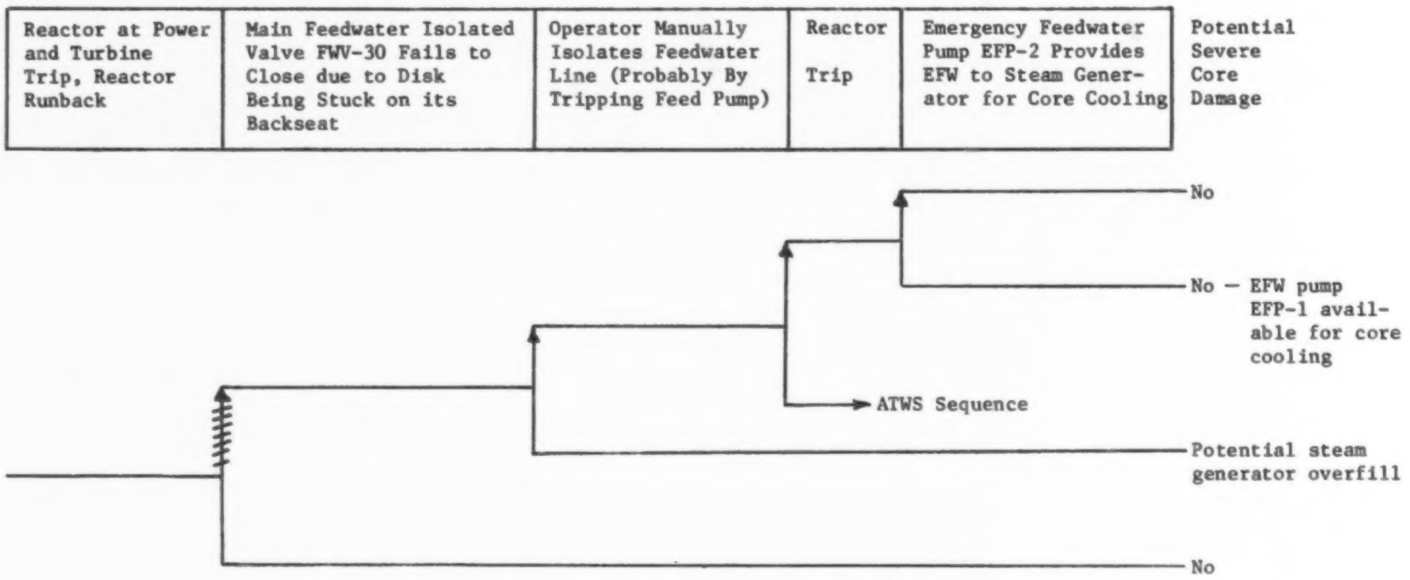
1. The valve is part of the feedwater flow control system. It is also used to isolate the steam generator from the feed system when required.

Unavailability of system per WASH 1400:* —

Unavailability of component per WASH 1400:* valve, MOV, failure to operate: $1 \times 10^{-3}/D$

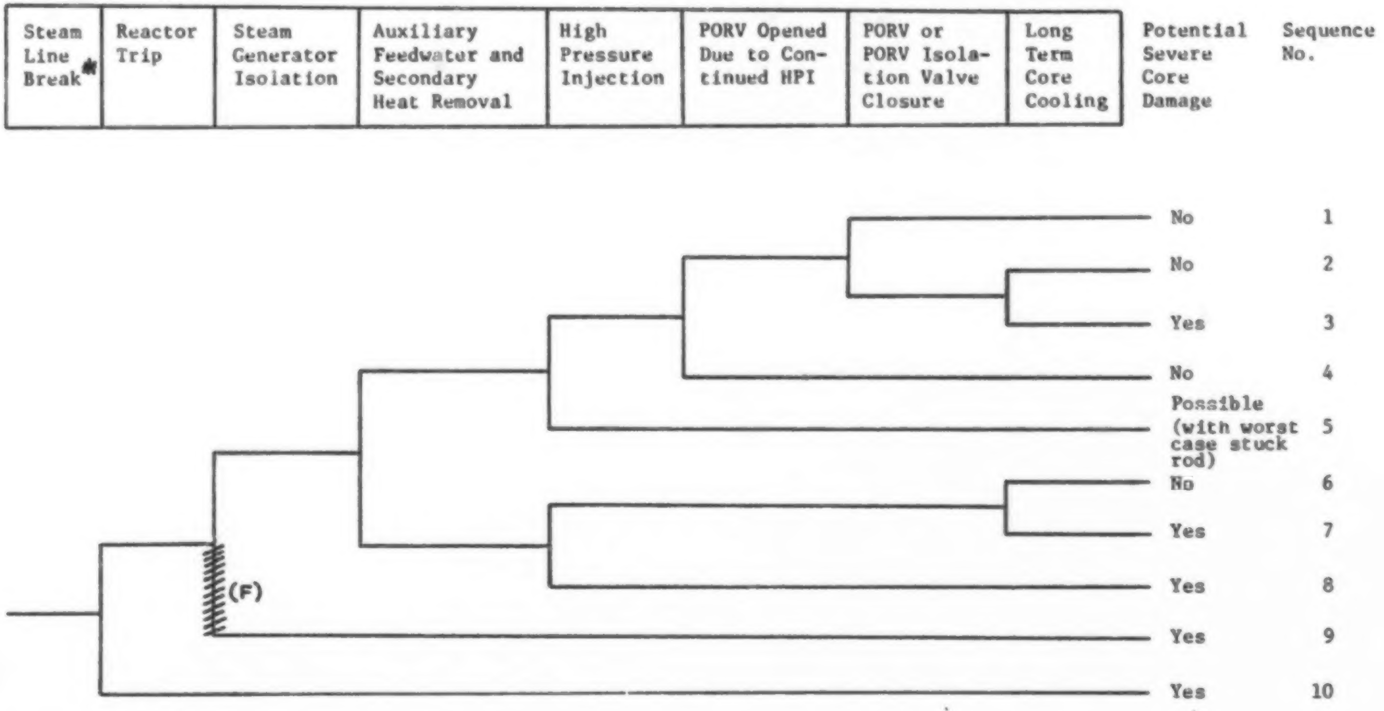
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor at Power and Turbine Trip, Reactor Runback
Main Feedwater Isolated Valve FWV-30 Fails to Close due to Disk Being Stuck on its Backseat
Operator Manually Isolates Feedwater Line (Probably By Tripping Feed Pump)
Reactor Trip
Emergency Feedwater Pump EFP-2 Provides EFW to Steam Generator for Core Cooling
Potential Severe Core Damage
No
No — EFW pump EFP-1 available for core cooling
ATWS Sequence
Potential steam generator overfill
No

NSIC 147223 — Actual Occurrence for Feedwater Block Valve Fails to Close at Crystal River 3

579


Steam Line Break*
Reactor Trip
Steam Generator Isolation
Auxiliary Feedwater and Secondary Heat Removal
High Pressure Injection
PORV Opened Due to Continued HPI
PORV or PORV Isolation Valve Closure
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
(F)
No 1
No 2
Yes 3
No 4
Possible (with worst case stuck rod) 5
No 6
Yes 7
Yes 8
Yes 9
Yes 10

NSIC 147223 — Sequence of Interest for Feedwater Block Valve Fails to Close at Crystal River 3

*"A" Steam Generator Steam Line

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: **147223**

DATE OF LER: **January 26, 1979**

DATE OF EVENT: **January 6, 1979**

SYSTEM INVOLVED: **feedwater**

COMPONENT INVOLVED: **valve**

CAUSE: **Stuck on its back seat**

SEQUENCE OF INTEREST: **steam line break**

ACTUAL OCCURRENCE: **turbine trip and stuck FW valve**

REACTOR NAME: **Crystal River 3**

DOCKET NUMBER: **50-302**

REACTOR TYPE: **PWR**

DESIGN ELECTRICAL RATING: **825** MWe

REACTOR AGE: **2.0** yr

VENDOR: **B&W**

ARCHITECT-ENGINEERS: **Gilbert Associates**

OPERATORS: **Florida Power Corp.**

LOCATION: **7 miles NW of Crystal River, Fla.**

DURATION: 360(a) hours

PLANT OPERATING CONDITION: **45% power**

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; (c) made inoperable; [✓ (d) failed to operate.]

DISCOVERY METHOD: **operational event**

COMMENT: **On reactor runback after turbine trip, main FW valve failed to close.**

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 147383

Date: February 23, 1979

Title: Both Diesel Generators Inoperable Twice at Cook 1

The failure sequence was:

First Incident — With the reactor at 100% power, one diesel generator was running in parallel, which requires its undervoltage bus stripping capability to be blocked. While the first diesel generator was running, the second was removed from service for maintenance.

Second Incident — With the reactor at 100% power, one diesel generator was inoperable because of maintenance. The second diesel generator was started and its field failed to excite.

Corrective action:

First Incident — None

Second Incident — The generator field was energized and then depowered after the diesel generator was shut down. After this, the field successfully flashed using residual magnetism when the diesel was started.

Design purpose of failed system or component:

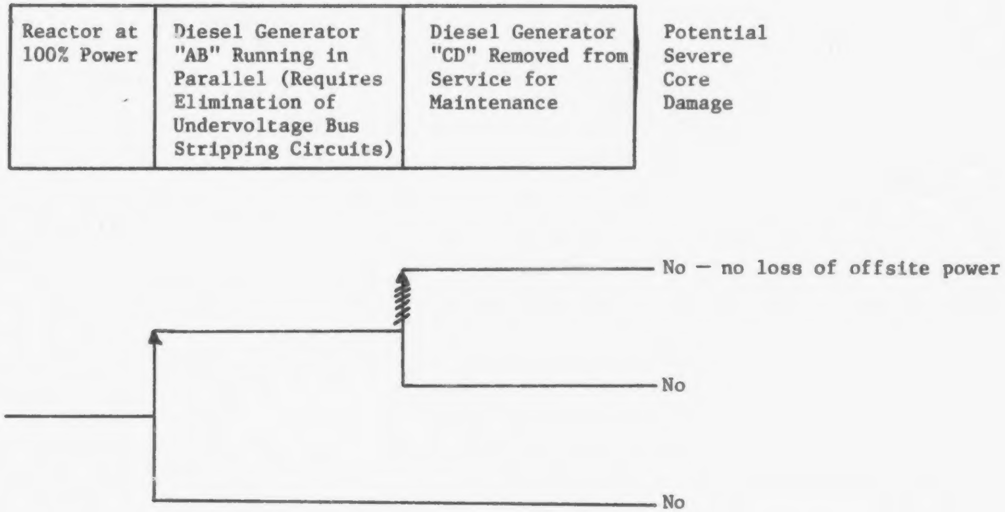
The diesel generators provide an on-site source of power to safety-related components when the unit generator and offsite sources are not available.

Unavailability of system per WASH 1400:* emergency power: 1×10^{-2}/D

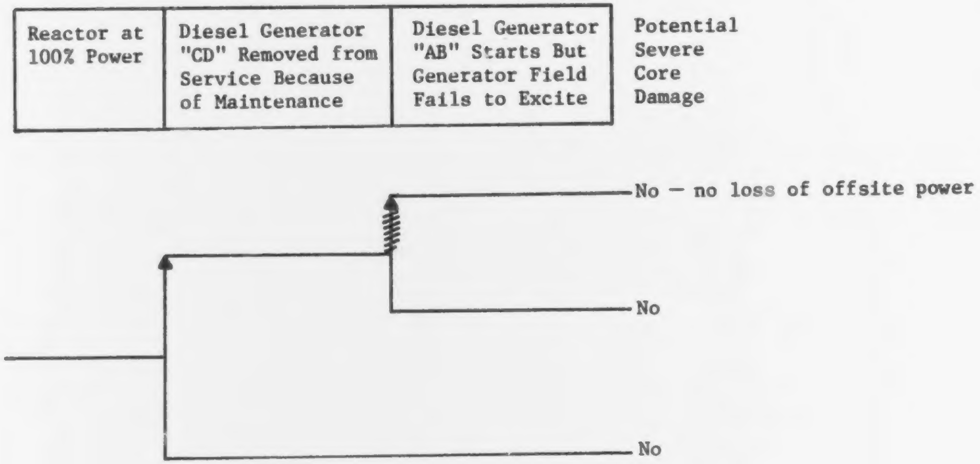
Unavailability of component per WASH 1400:* diesel generator: 3×10^{-2}/D

---

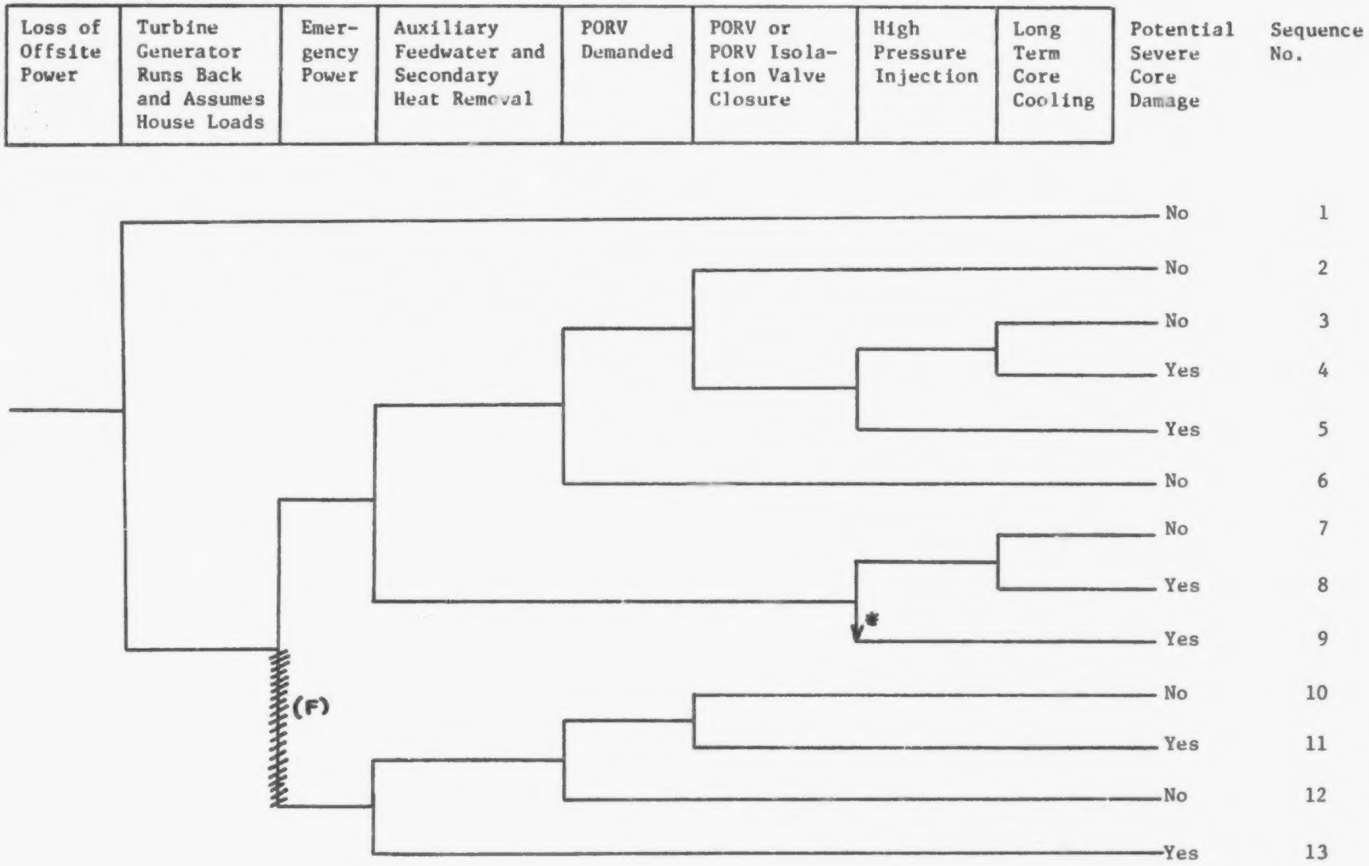
*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor at 100% Power
Diesel Generator "AB" Running in Parallel (Requires Elimination of Undervoltage Bus Stripping Circuits)
Diesel Generator "CD" Removed from Service for Maintenance
Potential Severe Core Damage
No — no loss of offsite power
No
No

NSIC 147383 — Actual Occurrence for Both Diesel Generators Inoperable Twice at Cook 1 (First Incident)


Reactor at 100% Power
Diesel Generator "CD" Removed from Service Because of Maintenance
Diesel Generator "AB" Starts But Generator Field Fails to Excite
Potential Severe Core Damage
No — no loss of offsite power
No
No

NSIC 147383 — Actual Occurrence for Both Diesel Generators Inoperable Twice at Cook 1 (Second Incident)


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
Yes 9
No 10
Yes 11
No 12
Yes 13
(F)
*

NSIC 147383 — Sequence of Interest for Both Diesel Generators Inoperable Twice at Cook 1

*not included in mitigation procedures.

585

BEST DOCUMENT AVAILABLE

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 147383

DATE OF LER: March 13, 1979

DATE OF EVENT: February 23, 1979

SYSTEM INVOLVED: Emergency Power

COMPONENT INVOLVED: Diesel Generators

CAUSE: Maintenance on Two Diesel Generators at One Time (First Incident); Failure of One Diesel Generator to Excite its Field While the Other Generator was Unavailable Due to Maintenance.

SEQUENCE OF INTEREST: Loss of Offsite Power

ACTUAL OCCURRENCE: Unavailability of Two Diesel Generators (Two Occasions)

REACTOR NAME: Cook 1

DOCKET NUMBER: 50-315

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 1054 MWe

REACTOR AGE: 4.1 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: American Electric Power Service Corp.

OPERATORS: Indiana and Michigan Electric Co.

LOCATION: 11 miles south of Benton Harbor, Michigan

DURATION: 1.1 hours

PLANT OPERATING CONDITION: at 100% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; [✓ (b) failed to start]; [✓ (c) made inoperable]; (d) ____________________

DISCOVERY METHOD: Operator observation

COMMENT: A 1979 revision to the Cook 1 & 2 FSAR includes a change to the turbine driven AFW pump start and control circuitry to power this circuitry from a separate 250VDC supply. This will eliminate the dependency between the diesel generators and the turbine driven AFW pump and permit operation of the turbine driven AFW pump w/o the diesel generators. This will provide reactor cooling even if both diesel generators fail to start or run.

BEST DOCUMENT AVAILABLE

586

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 147399

Date: March 22, 1979

Title: Both Diesel Generators Unavailable at Kewaunee

The failure sequence was:

1. With the reactor at full power, diesel generator 1B was removed from service for maintenance.
2. During the testing of diesel 1A, manual speed control was lost due to excessive wear to the bronze pinion in the syncho-motor gear housing. The pinion damage was caused by improper gear alignment and failure of a governor limit switch.

Corrective action:

The pinion and limit switch were replaced and adjusted and the diesel generator was tested and returned to service.

Design purpose of failed system or component:

The diesel generators provide power to safety-related equipment when offsite power and the unit generator are unavailable.

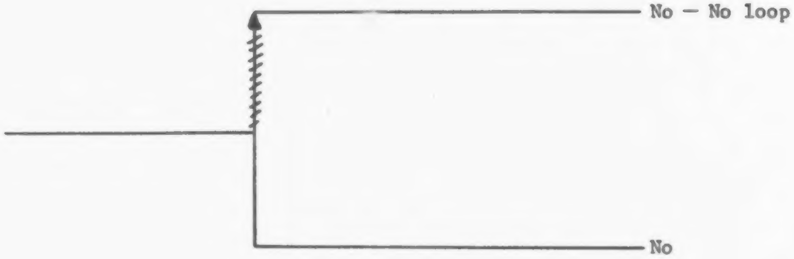
Unavailability of system per WASH 1400:* Emergency power: 1×10^{-2}/D

Unavailability of component per WASH 1400:* Diesel generator: 3×10^{-2}/D

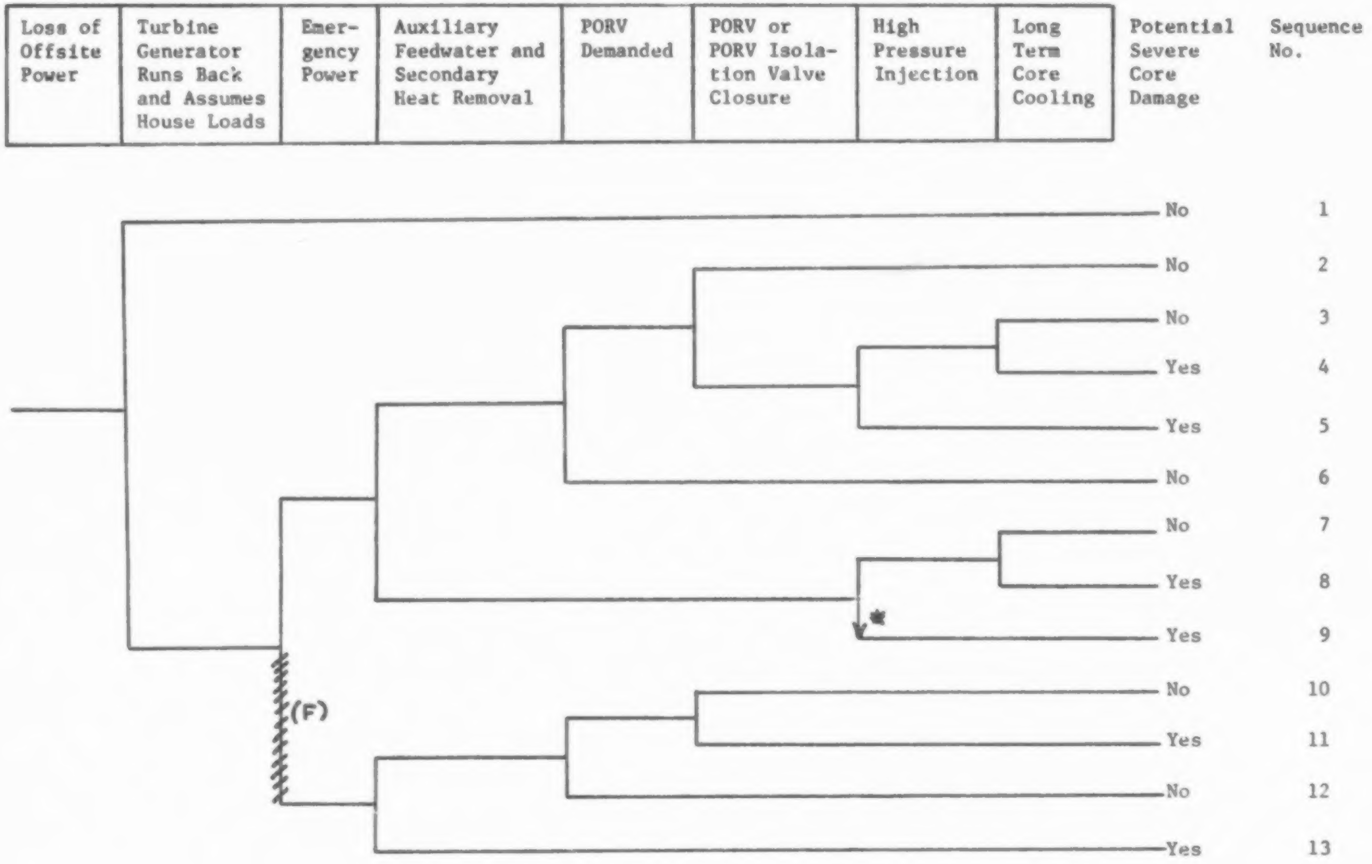
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

Reactor at Power and Diesel Generator 1B Removed From Service for Maintenance	Diesel Generator 1A Fails to Run Due to Loss of Manual Speed Control — A Result of a Worm Pinion in the Synchro-Motor Gear Housing and a Failed Limit Switch	Potential Severe Core Damage



NSIC 147399 — Actual Occurrence for Both Diesel Generators Unavailable at Kewaunee


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
Yes 9
No 10
Yes 11
No 12
Yes 13
(F)
*

NSIC 147399 — Sequence of Interest for Both Diesel Generators Unavailable at Kewaunee

*Not included in operating procedures

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 147399

DATE OF LER: March 22, 1979

DATE OF EVENT: February 21, 1979

SYSTEM INVOLVED: Emergency power

COMPONENT INVOLVED: Diesel generator

CAUSE: Excessive wear on pinion in synchro-gear motor housing plus failed limit switch

SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE: Unavailability of two diesel generators while at power

REACTOR NAME: Kewaunee

DOCKET NUMBER: 305

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 535 MWe

REACTOR AGE: 5.0 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Flour Power

OPERATORS: Wisconsin Public Service Corp.

LOCATION: 27 miles East of Green Bay, Wisconsin

DURATION: 4(a) hours

PLANT OPERATING CONDITION: 100% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; (c) made inoperable; [✓ (d)] failed to run

DISCOVERY METHOD: Testing

COMMENT:

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 147400

Date: February 19, 1979

Title: Both Main Steam Stop Valves Fail to Close at Point Beach 2

The failure sequence was:

1. With the reactor at 10% power and main steam stop valve testing in progress, both main steam stop valves failed to close.

Corrective action:

The valves were eventually closed with mechanical assistance. Both valves were stroked several times and eventually functioned without assistance.

Design purpose of failed system or component:

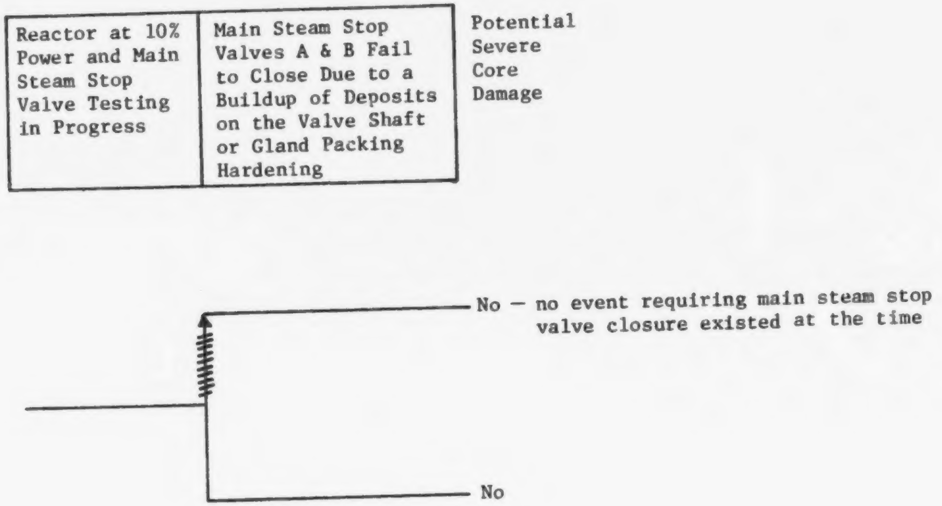
The main steam stop valves serve to isolate the steam generators from the main steam lines and prevent multiple steam generator blowdown during certain steam line break accidents.

Unavailability of system per WASH 1400:* not considered in WASH-1400.

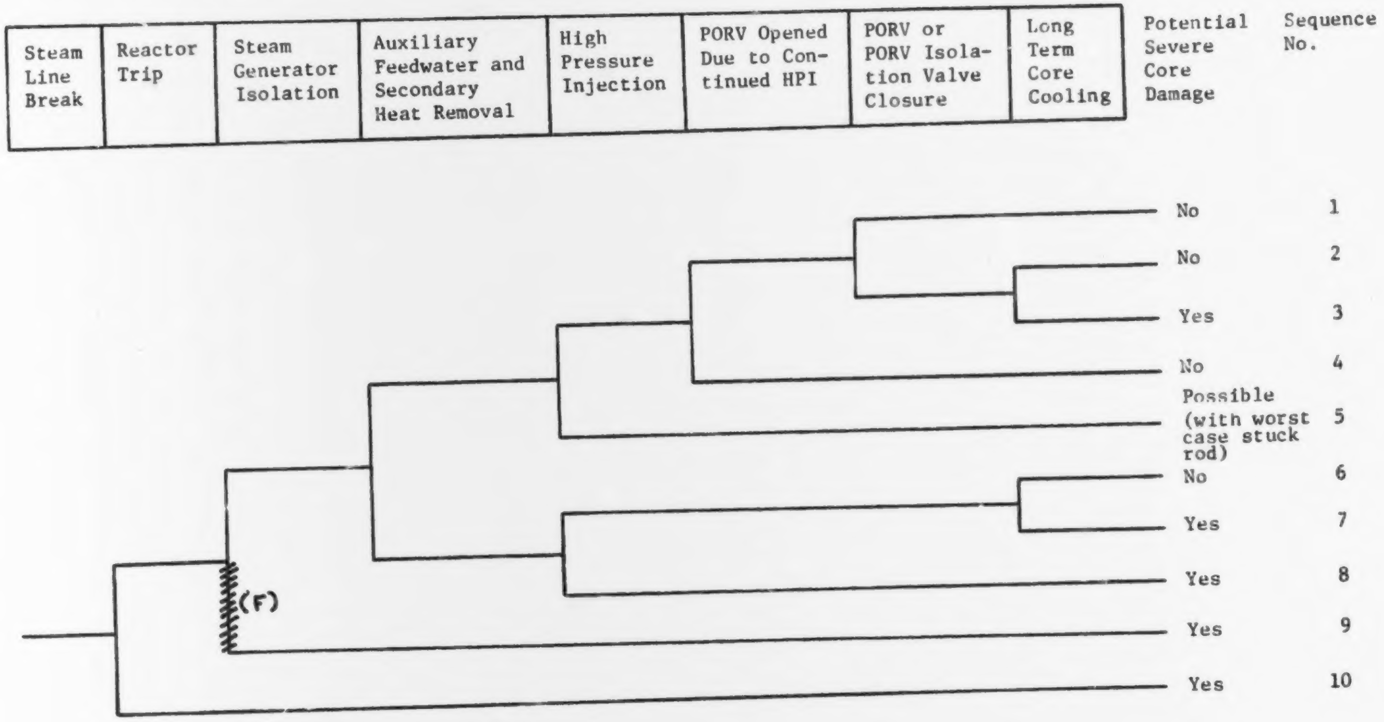
Unavailability of component per WASH 1400:* specific type not considered in WASH-1400

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}


Reactor at 10% Power and Main Steam Stop Valve Testing in Progress
Main Steam Stop Valves A & B Fail to Close Due to a Buildup of Deposits on the Valve Shaft or Gland Packing Hardening
Potential Severe Core Damage
No — no event requiring main steam stop valve closure existed at the time
No

NSIC 147400 — Actual Sequence for Both Main Steam Stop Valves Fail to Close at Point Beach 2


Steam Line Break
Reactor Trip
Steam Generator Isolation
Auxiliary Feedwater and Secondary Heat Removal
High Pressure Injection
PORV Opened Due to Continued HPI
PORV or PORV Isolation Valve Closure
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
(F)
No 1
No 2
Yes 3
No 4
Possible (with worst case stuck rod) 5
No 6
Yes 7
Yes 8
Yes 9
Yes 10

NSIC 147400 — Sequence of Interest for Both Main Steam Stop Valves Fail to Close at Point Beach 2

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 147400

DATE OF LER: February 23, 1979

DATE OF EVENT: February 19, 1979

SYSTEM INVOLVED: Main steam

COMPONENT INVOLVED: Main steam isolation valves

CAUSE: Valve failure to close due to deposits on valve shaft or packing gland hardening.

SEQUENCE OF INTEREST: Steam line break

ACTUAL OCCURRENCE: Main steam stop valves failure to close during testing

REACTOR NAME: Point Beach 2

DOCKET NUMBER: 50-301

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 497 MWe

REACTOR AGE: 6.7 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Wisconsin Electric Power Co.

LOCATION: 15 miles north of Manitowoc, Wisconsin

DURATION: 360(a) hours

PLANT OPERATING CONDITION: 10% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; (c) made inoperable; [✓ (d)] failed to close

DISCOVERY METHOD: During testing

COMMENT: –

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 148764

Date: January 18, 1979

Title: Steam Dump Valves Fail to Close Following Load Rejection at Beaver Valley 1

The failure sequence was:

1. With the reactor at 92% power, the 1B heater drain pump tripped due to low heater drain tank level.
2. During the subsequent load rejection, the condenser steam dump valves opened as required, but failed to close because the valve operators were exposed to cold temperatures due to improperly positioned air dampers in the turbine building.
3. Low steam line pressure resulted in a reactor trip and safety injection.

Corrective action:

The air dampers were properly positioned.

Design purpose of failed system or component:

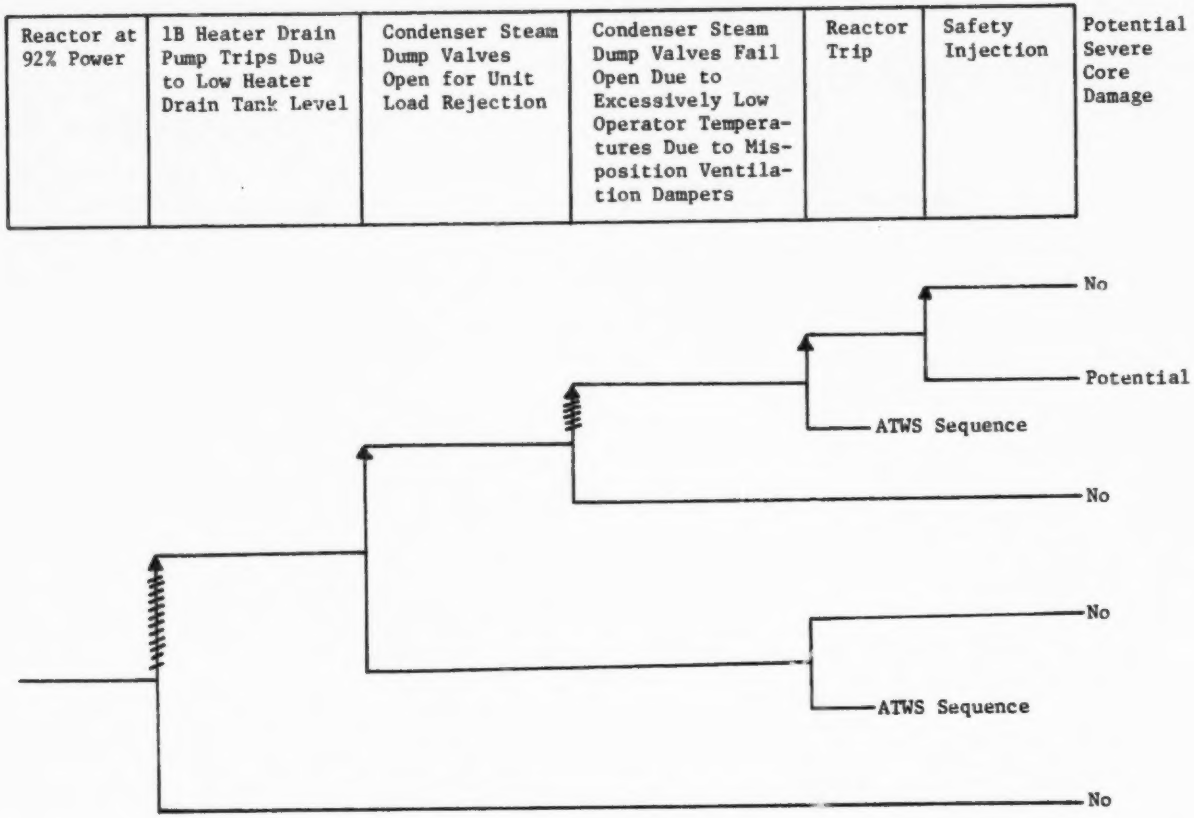
The condenser steam dump valves direct steam flow from the steam generators directly to the main condenser in the event of a loss of load or turbine trip, permitting reactor runback in lieu of trip.

Unavailability of system per WASH 1400:* not addressed in WASH-1400.

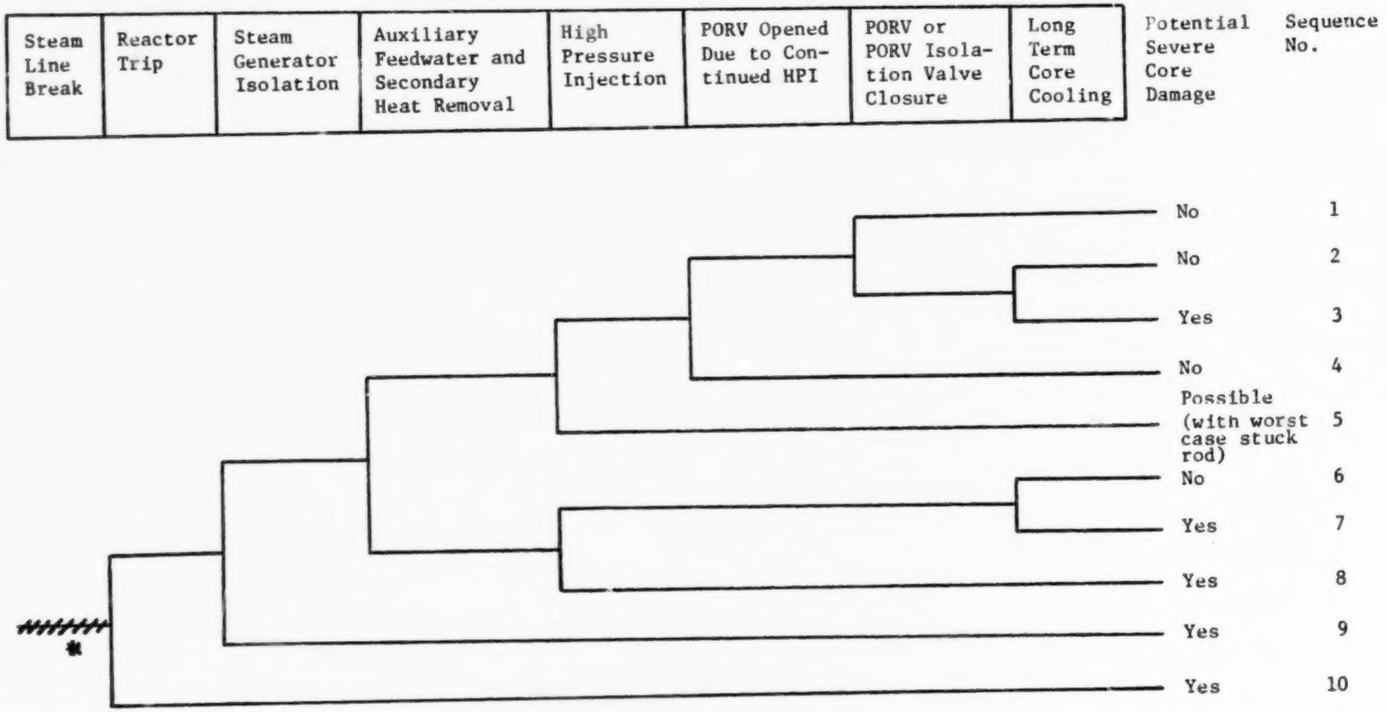
Unavailability of component per WASH 1400:* motor-operated valve, failure to operate: 1×10^{-3}/D.

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor at 92% Power
1B Heater Drain Pump Trips Due to Low Heater Drain Tank Level
Condenser Steam Dump Valves Open for Unit Load Rejection
Condenser Steam Dump Valves Fail Open Due to Excessively Low Operator Temperatures Due to Misposition Ventilation Dampers
Reactor Trip
Safety Injection
Potential Severe Core Damage
No
Potential
ATWS Sequence
No
No
ATWS Sequence
No

NSIC 148764 — Actual Sequence for Steam Dump Valves Fail to Close Following Load Rejection at Beaver Valley


Steam Line Break
Reactor Trip
Steam Generator Isolation
Auxiliary Feedwater and Secondary Heat Removal
High Pressure Injection
PORV Opened Due to Continued HPI
PORV or PORV Isolation Valve Closure
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
Yes 3
No 4
Possible (with worst case stuck rod) 5
No 6
Yes 7
Yes 8
Yes 9
Yes 10
*

NSIC 148764 — Sequence of Interest for Steam Dump Valves Fail to Close Following Load Rejection at Beaver Valley 1

*small

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 148764

DATE OF LER: April 18, 1979

DATE OF EVENT: January 18, 1979

SYSTEM INVOLVED: Main steam

COMPONENT INVOLVED: Condenser steam dump valves

CAUSE: Valves failed to close due to exposure to abnormally low temperatures resulting from improper damper position, human error

SEQUENCE OF INTEREST: Steam line break

ACTUAL OCCURRENCE: Stuck open steam dump valves

REACTOR NAME: Beaver Valley 1

DOCKET NUMBER: 50-334

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 852 MWe

REACTOR AGE: 2.7 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Stone & Webster

OPERATORS: Duquesne Light Co.

LOCATION: 5 miles east of East Liverpool, Ohio

DURATION: N/A

PLANT OPERATING CONDITION: 92% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; (c) made inoperable; [✓ (d)] failed to close

DISCOVERY METHOD: Operator observation, SI initiation

COMMENT: The LER does not specify how many of these valves stuck open. However, the FSAR (section 10.3) states that all the steam dump valves open on a step load decrease and the failure of 9 remaining open is equivalent to a steam line break.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 148892

Date: February 22, 1979

Title: Diesel Generator Sequencer Found in Test Mode at Farley 1

The failure sequence was:

1. With the plant at 94% power, diesel generator 1B was removed from service for preventive maintenance.
2. The 1-2A diesel generator sequencer was found to be in the test mode, rendering that diesel generator inoperable. The sequencer was in the test mode due to a faulty switch mechanism.

Corrective action:

The sequencer was returned to operation and the faulty switch mechanism repaired.

Design purpose of failed system or component:

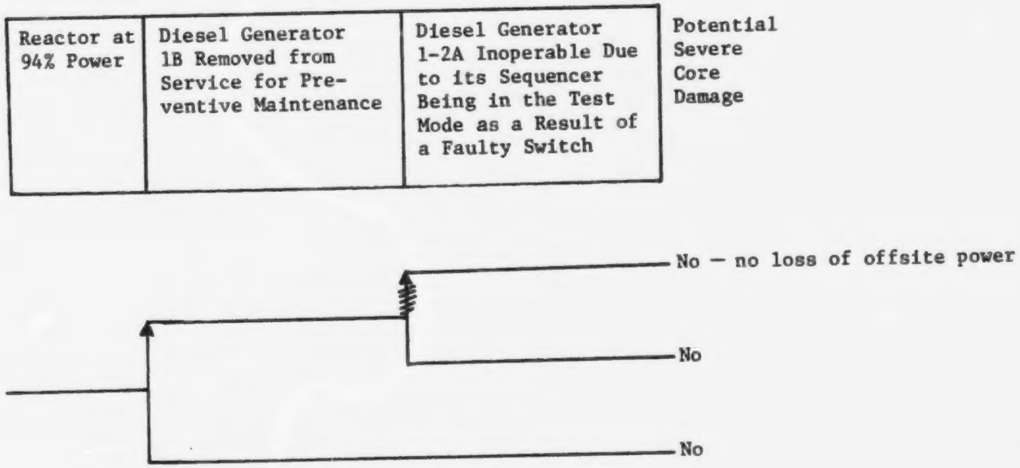
The sequencer provides control signals for load shedding and load sequencing in connection with diesel generator starting and loading.

Unavailability of system per WASH 1400:* Electric power: 1×10^{-2}/D

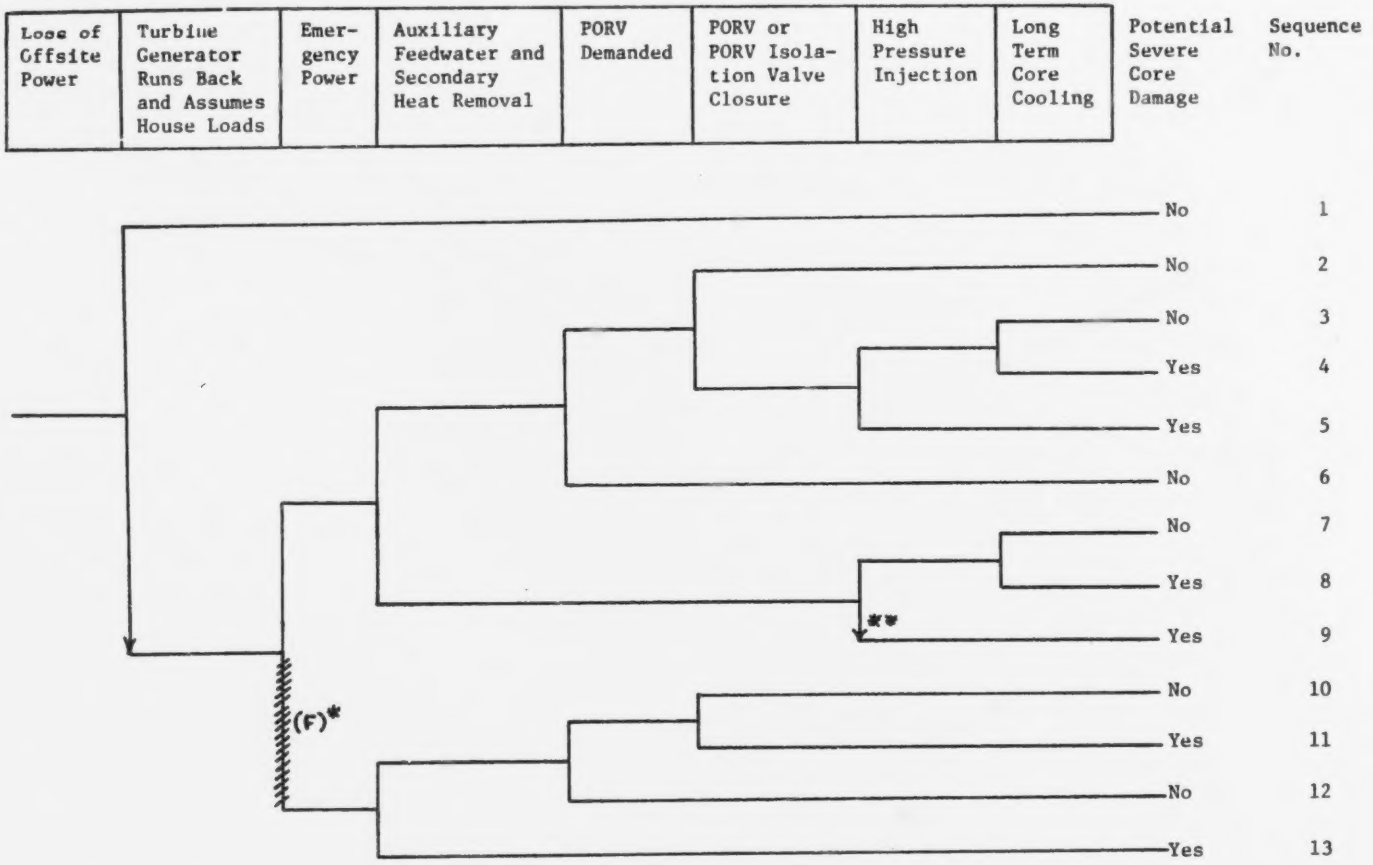
Unavailability of component per WASH 1400:* Diesel generator: 3×10^{-2}/D

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor at 94% Power
Diesel Generator 1B Removed from Service for Preventive Maintenance
Diesel Generator 1-2A Inoperable Due to its Sequencer Being in the Test Mode as a Result of a Faulty Switch
Potential Severe Core Damage
No — no loss of offsite power
No
No

NSIC 148892 — Actual Occurrence for Diesel Generator Sequencer Found in Test Mode at Farley 1


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
Yes 9
No 10
Yes 11
No 12
Yes 13
(F)*
**

NSIC — 148892 — Sequence of Interest for Diesel Generator Sequencer Found in Test Mode at Farley 1

*success requires operator action to transfer loads to a smaller, river water pump diesel generator.

**not included in mitigation procedures.

BEST DOCUMENT AVAILABLE

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 148892

DATE OF LER: March 22, 1979

DATE OF EVENT: February 22, 1979

SYSTEM INVOLVED: Emergency power

COMPONENT INVOLVED: Diesel generator sequencer

CAUSE: Sequencer in test mode due to faulty switch

SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE: Sequencer in test while other diesel generator was unavailable

REACTOR NAME: Farley 1

DOCKET NUMBER: 50-348

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 829 MWe

REACTOR AGE: 1.5 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Bechtel/SSI

OPERATORS: Alabama Power Co.

LOCATION: 24 miles SE of Dothan, Alabama

DURATION: 24(a) hours

PLANT OPERATING CONDITION: 94% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c) made inoperable]; (d) ____________________

DISCOVERY METHOD: Operator observation

COMMENT: —

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: **149230**

Date: **March 21, 1979**

Title: **Both Emergency Service Water Systems Inoperable at Arnold**

The failure sequence was:

1. **The reactor was operating at 70% power and surveillance testing of the emergency service water system was underway.**
2. **Strainer in both emergency service water systems were found to be plugged.**

Corrective action:

1. **The strainers were cleaned within 2.5 hours.**
2. **Self-cleaning strainers are to be installed during the 1980 refuel.**

Design purpose of failed system or component:

The emergency service water system provides cooling water to the following systems given a loss of offsite power:

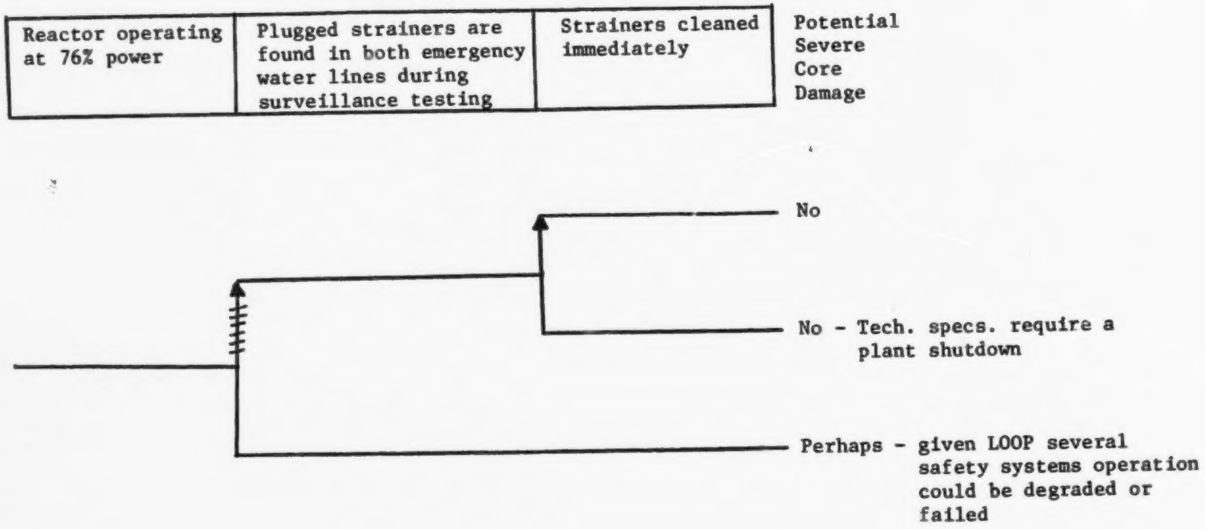
1. **diesel generators**
2. **RHR pump seal coolers**
3. **RHR and core spray room cooling unit**
4. **HPCI room cooling unit**
5. **RCIC room cooling unit**
6. **control building chiller**
7. **core spray pump motor bearing coolers**

Unavailability of system per WASH 1400:* **not given explicitly**

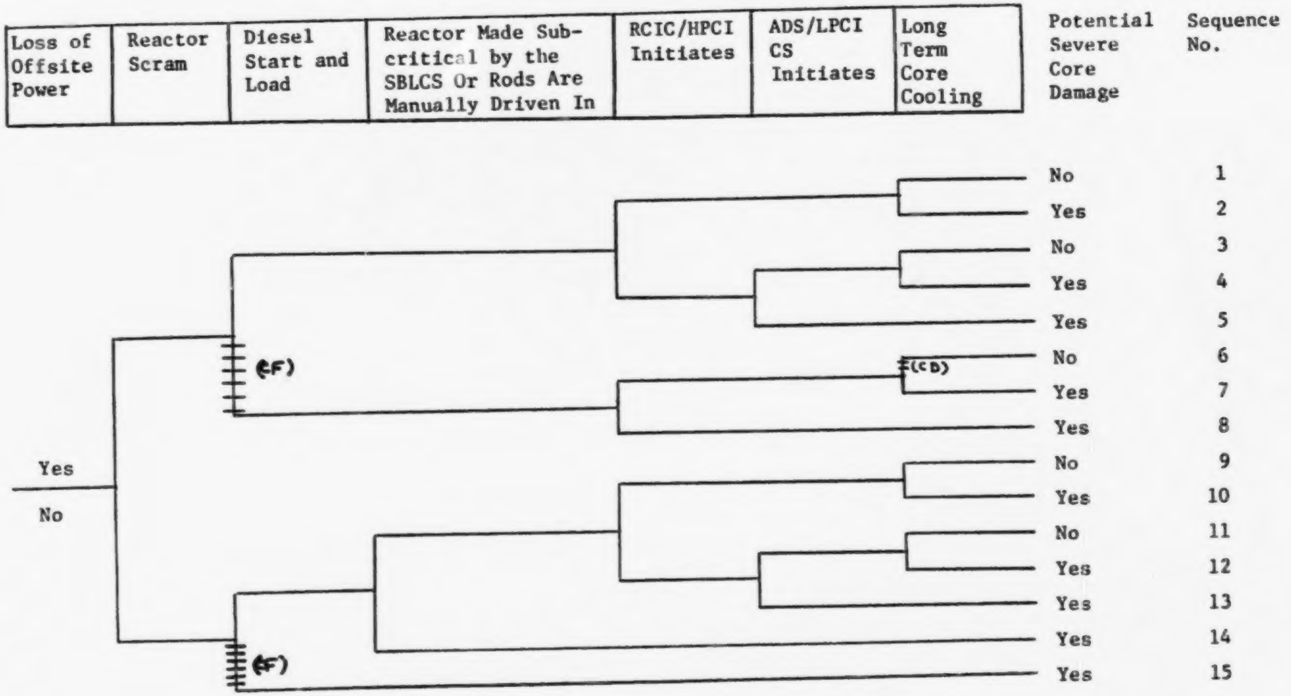
Unavailability of component per WASH 1400:* -

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor operating at 76% power
Plugged strainers are found in both emergency water lines during surveillance testing
Strainers cleaned immediately
Potential Severe Core Damage
No
No - Tech. specs. require a plant shutdown
Perhaps - given LOOP several safety systems operation could be degraded or failed

NSIC 149230 — Actual Occurrence of Both Emergency Service Water Systems Inoperable at Arnold


Loss of Offsite Power
Reactor Scram
Diesel Start and Load
Reactor Made Sub-critical by the SBLCS Or Rods Are Manually Driven In
RCIC/HPCI Initiates
ADS/LPCI CS Initiates
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
(F)
(CD)
(F)
No 1
Yes 2
No 3
Yes 4
Yes 5
No 6
Yes 7
Yes 8
No 9
Yes 10
No 11
Yes 12
Yes 13
Yes 14
Yes 15

NSIC 149230 — Sequence of Interest of Both Emergency Service Water Systems Inoperable at Arnold

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 149230

DATE OF LER: April 4, 1979

DATE OF EVENT: March 21, 1979

SYSTEM INVOLVED: emergency service water system

COMPONENT INVOLVED: strainers

CAUSE: the strainers were not cleaned often enough

SEQUENCE OF INTEREST: loss of offsite power

ACTUAL OCCURRENCE: both emergency service water systems inoperable at Arnold

REACTOR NAME: Arnold

DOCKET NUMBER: 50-331

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 538 MWe

REACTOR AGE: 5.0 yr

VENDOR: GE

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Iowa Light and Power

LOCATION: 8 miles NW of Cedar Rapids

DURATION: 360(a) hours

PLANT OPERATING CONDITION: 76%

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________________

DISCOVERY METHOD: surveillance testing

COMMENT: -

BEST DOCUMENT AVAILABLE

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 149250

Date: April 14, 1979

Title: **Two Main Steam Isolation Valves Fail to Close at Trojan**

The failure sequence was:

1. **With the reactor in hot standby, main steam isolation valve testing was in progress.**
2. **Two of the main steam isolation valves failed to close when manually initiated due to binding between the valve stem and packing.**

Corrective action:

1. **The valve packings were adjusted so the valves operated freely.**

Design purpose of failed system or component:

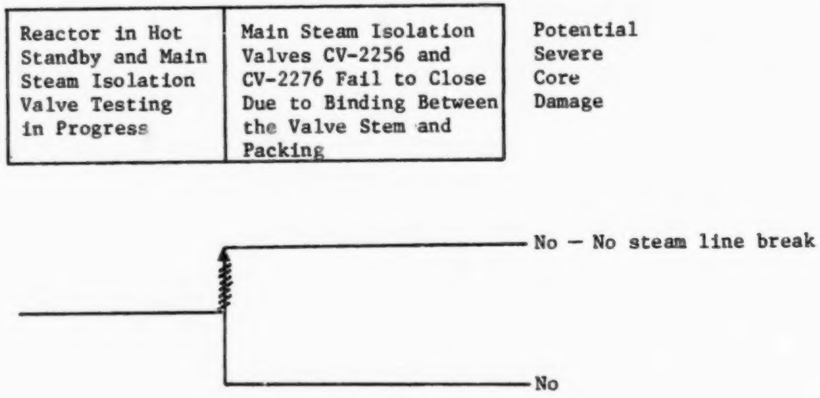
The main steam isolation valves isolate the steam generators during a steam line break, and prevent blowdown of more than one steam generator.

Unavailability of system per WASH 1400:* —

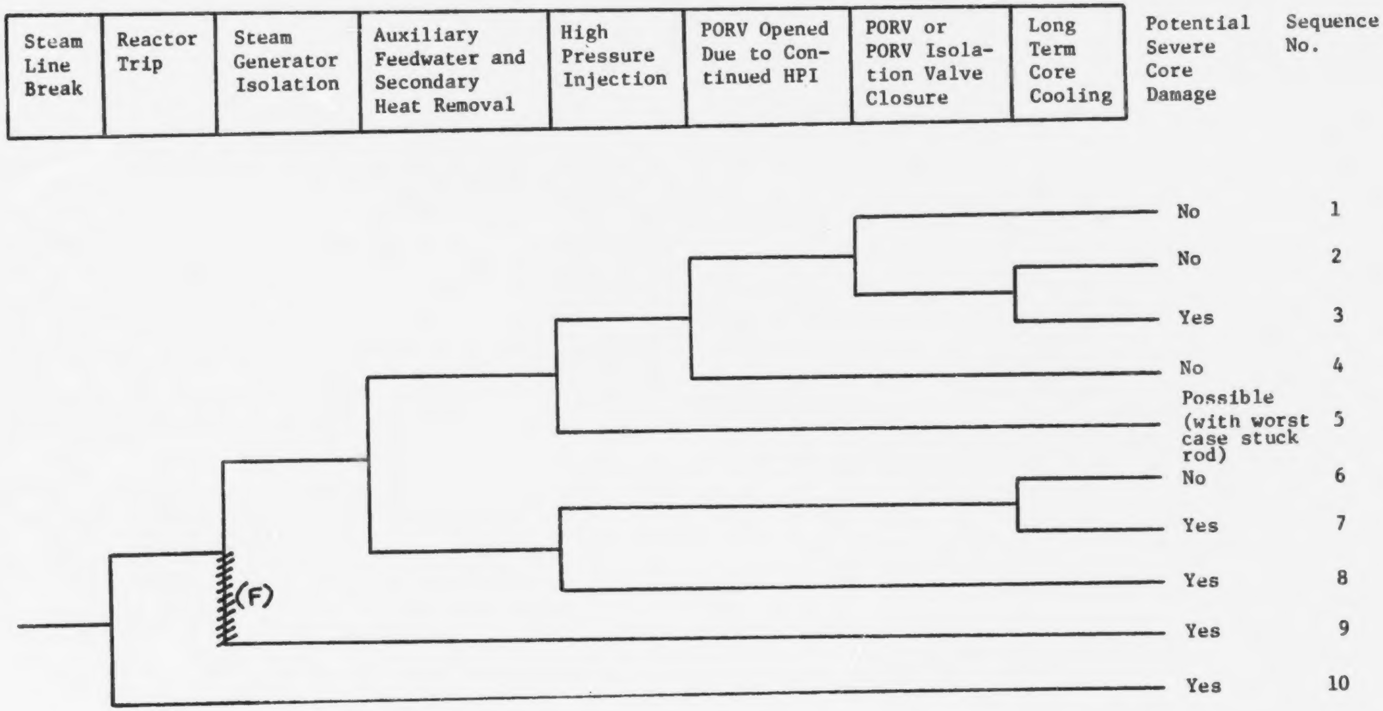
Unavailability of component per WASH 1400:* **valve, failure to operate:** 10^{-3}

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor in Hot Standby and Main Steam Isolation Valve Testing in Progress
Main Steam Isolation Valves CV-2256 and CV-2276 Fail to Close Due to Binding Between the Valve Stem and Packing
Potential Severe Core Damage
No — No steam line break
No

NSIC 149250 — Actual Occurrence for Two Main Steam Isolation Valves Fail to Close at Trojan


Steam Line Break
Reactor Trip
Steam Generator Isolation
Auxiliary Feedwater and Secondary Heat Removal
High Pressure Injection
PORV Opened Due to Continued HPI
PORV or PORV Isolation Valve Closure
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
(F)
No 1
No 2
Yes 3
No 4
Possible (with worst case stuck rod) 5
No 6
Yes 7
Yes 8
Yes 9
Yes 10

NSIC 149250 — Sequence of Interest for Two Main Steam Isolation Valves Fail to Close at Trojan

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 149250

DATE OF LER: April 27, 1979

DATE OF EVENT: April 14, 1979

SYSTEM INVOLVED: Main steam

COMPONENT INVOLVED: Main steam isolation valves

CAUSE: Valves failed to close due to binding between valve stem and packing.

SEQUENCE OF INTEREST: Steam line break

ACTUAL OCCURRENCE: Valve failed to close during testing.

REACTOR NAME: Trojan

DOCKET NUMBER: 50-344

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 1130 MWe

REACTOR AGE: 3.3 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Portland General Electric Co.

LOCATION: 42 miles north of Portland, Oregon

DURATION: 360(a) hours

PLANT OPERATING CONDITION: Hot standby

SAFETY FEATURE TYPE OF FAILURE: [✓ (a)] inadequate performance; (b) failed to start; (c) made inoperable; (d) ______________________

DISCOVERY METHOD: During testing.

COMMENT: —

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: **149450**

Date: **May 2, 1979**

Title: **Loss of Feedwater Transient at Oyster Creek**

The failure sequence was:

1. A spurious high RCS pressure signal caused by surveillance testing caused a reactor scram and an ATWS recirculation pump trip.
2. The turbine/generator tripped due to low load. This caused all three reactor feed pumps to trip.
3. An automatic transfer was made to the "SA" startup transformer and buses "1A" and "1C" loaded. "SB" startup transformer was out of service for maintenance thus a transfer did not take place. Diesel generator #2 started and loaded bus "1D", however power was lost to non-essential bus "1B".
4. An unsuccessful attempt to start the "A" feed pump was made. "B" and "C" feed pumps are feed from the "1B" bus which was de-energized.

(see attached page)

Corrective action:

1. The "A" feedwater pump and the "A" recirculation pump were placed into service and the water level returned to the proper level.
2. The reactor was placed in cold shutdown.

Design purpose of failed system or component:

The isolation condenser system prevents overheating of the reactor fuel in the event feedwater is lost and heat removal systems which require a.c. power for operation are unavailable.

Unavailability of system per WASH 1400:* not considered in WASH 1400

Unavailability of component per WASH 1400:* 3×10^{-3}/D, operator error

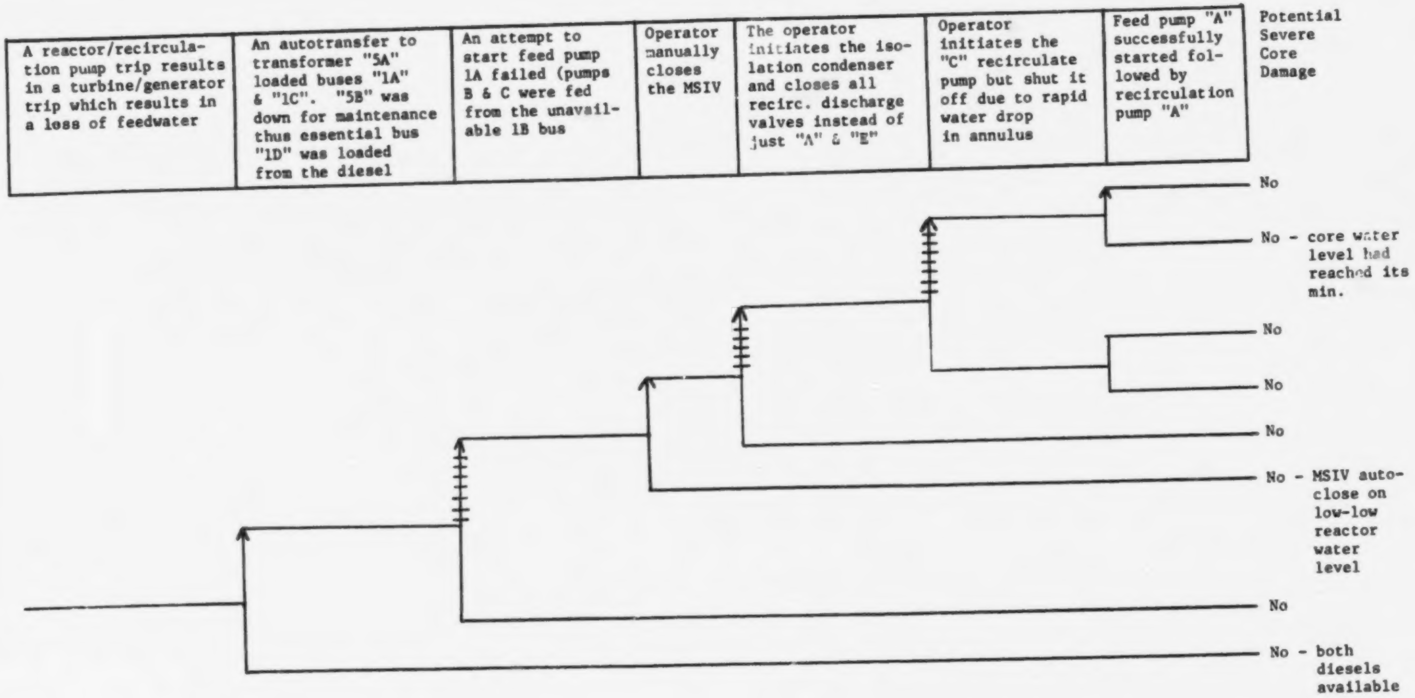
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}

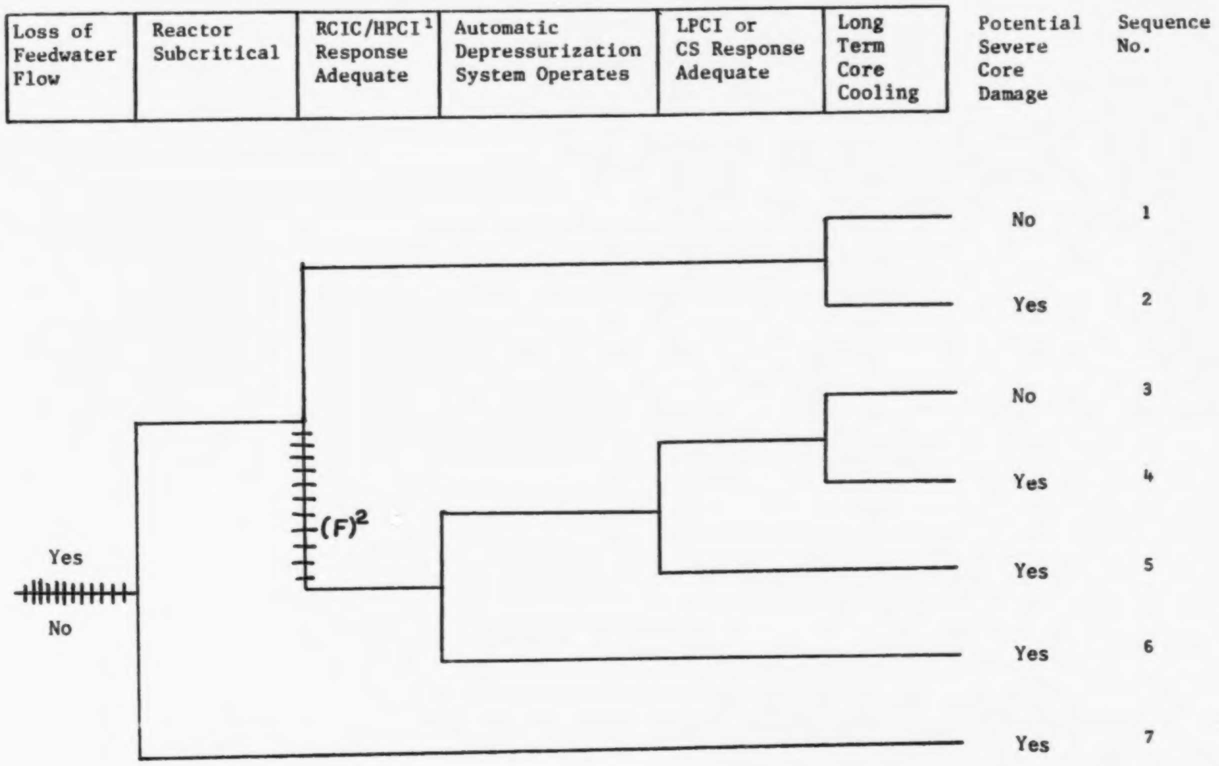
[redacted]

The failure sequence was: (continued)

5. The operator manually closed the MSIVs.
6. The "B" isolation condenser was manually initiated by the operator. This required closing discharge valves in the "A" and "E" recirculation loop. It is believed the operator also closed loops "B" and "C" discharge valves. The "D" loop had been isolated prior to the event for maintenance. This left all 5 recirculation loop discharge valves closed, thus isolating the core from returning condensate from the isolation condensers. All 5 bypass valves around the discharge valves were opened and once the "1D" bus was loaded both CRD pumps were operational, thus the core region received a continuous supply of water. Water from the isolation condenser backed up into the downcomer since flow to the core was very restricted.
7. The isolation condensers were operated periodically or to control the reactor cool down rate.
8. The operator initiated the "C" recirculation pump, however, it was quickly shut down due to a rapid water level drop in the downcomer annulus.
9. The "A" feedwater pump was finally initiated (after several unsuccessful attempts) thus providing high capacity, high pressure coolant injection. Shortly afterwards the "A" recirculation pump was started and water level in the reactor and annulus region achieved equilibrium.


A reactor/recirculation pump trip results in a turbine/generator trip which results in a loss of feedwater
An autotransfer to transformer "5A" loaded buses "1A" & "1C". "5B" was down for maintenance thus essential bus "1D" was loaded from the diesel
An attempt to start feed pump 1A failed (pumps B & C were fed from the unavailable 1B bus
Operator manually closes the MSIV
The operator initiates the isolation condenser and closes all recirc. discharge valves instead of just "A" & "E"
Operator initiates the "C" recirculate pump but shut it off due to rapid water drop in annulus
Feed pump "A" successfully started followed by recirculation pump "A"
Potential Severe Core Damage
No
No - core water level had reached its min.
No
No
No
No - MSIV auto-close on low-low reactor water level
No
No - both diesels available

NSIC 149450 – Actual Occurrence of a Loss of Feedwater Flow at Oyster Creek


Loss of Feedwater Flow
Reactor Subcritical
RCIC/HPCI¹ Response Adequate
Automatic Depressurization System Operates
LPCI or CS Response Adequate
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
(F)²
No 1
Yes 2
No 3
Yes 4
Yes 5
Yes 6
Yes 7

NSIC 149450 — Sequence of Interest for a Loss of Feedwater Flow at Oyster Creek

[1] Oyster Creek utilizes Isolation condensers rather than RCIC

[2] Success requires proper operation of either FWCT or IC.

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 149450

DATE OF LER: May 12, 1979

DATE OF EVENT: May 2, 1979

SYSTEM INVOLVED: isolation condensers

COMPONENT INVOLVED: recirculation loop discharge valves

CAUSE: operator/procedural deficiencies, human error

SEQUENCE OF INTEREST: loss of feedwater flow

ACTUAL OCCURRENCE: loss of feedwater flow at Oyster Creek

REACTOR NAME: Oyster Creek

DOCKET NUMBER: 50-219

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 650 MWe

REACTOR AGE: 10.4 yr

VENDOR: GE

ARCHITECT-ENGINEERS: Burns and Roe

OPERATORS: Jersey Central Power and Light

LOCATION: Nine miles S of Toms River, NJ

DURATION: N/A

PLANT OPERATING CONDITION: 98% power followed by scram

SAFETY FEATURE TYPE OF FAILURE: [✓ (a)] inadequate performance; (b) failed to start; (c) made inoperable; (d) ______________________

DISCOVERY METHOD: operation

COMMENT: Four previous LERs are attributed to problems with the I.C. system and its line rupture instrumentation. In fact, the administrative controls which require tripping the "A" and "C" recirculation pumps was developed to circumvent the line rupture problem. Perhaps if the problem with the instrumentation had been rectified instead of just the system this event could have bee avoided.

615

BEST DOCUMENT AVAILABLE

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: **149961**

Date: **June 3, 1979**

Title: **HPCI Fails to Start When Turbine Stop Valve Fails to Open at Hatch 2**

The failure sequence was:

1. **During power ascension the reactor tripped due to a condensate system trip.**
2. **HPCI failed to initiate on low-low water level due to a failed turbine stop valve. Water from HPCI mechanical seal cooling lines was leaking into the bracket drain cavity. The bracket drain cavity line had a valve installed for unknown reasons. This valve was closed which caused water to back up into the adjacent bearing housing and thus contaminate the pump oil.**
3. **The RCIC was out of service for unspecified reasons.**
4. **The reactor was placed in hot shutdown using the immediately repaired condensate system.**

Corrective action:

1. **The leaking seal line was replaced.**
2. **The oil system was drained, flushed and twice refilled.**
3. **The turbine bearing was inspected and found to be undamaged.**
4. **The valve in the bracket drain line was locked open and an investigation initiated to determine its significance.**

Design purpose of failed system or component:

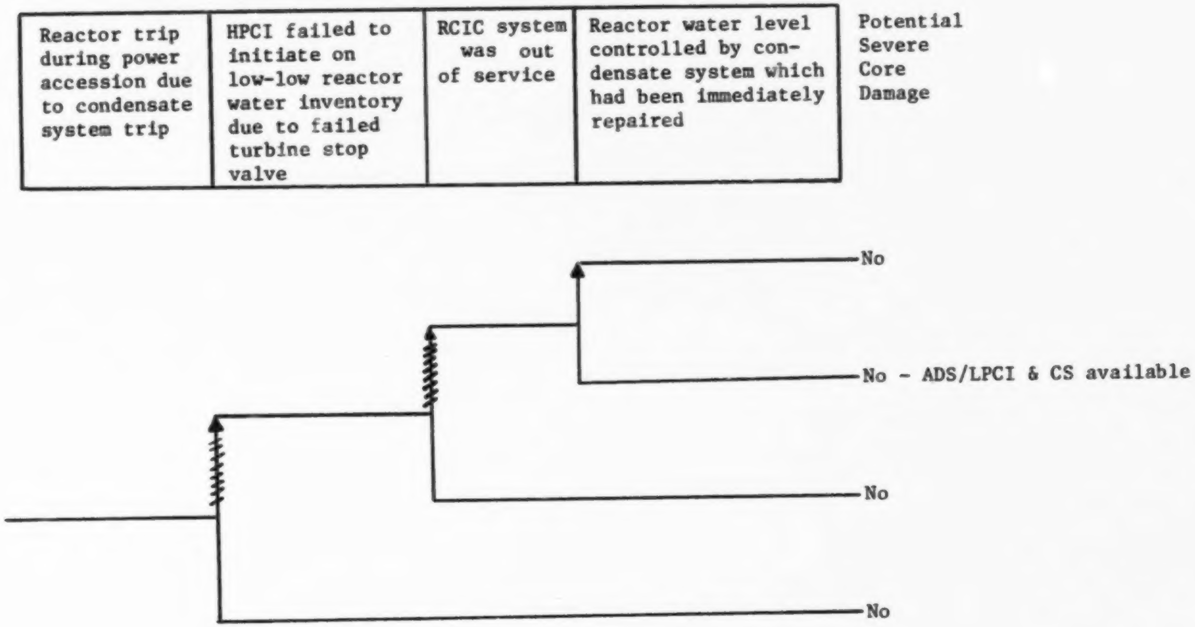
1. **The HPCI system is designed to control water level given a small LOCA.**
2. **The RCIC is designed to control water level given a loss of feedwater.**

Unavailability of system per WASH 1400:* **HPCI: 8.8×10^{-2}/D**
RCIC: 8.0×10^{-2}/D

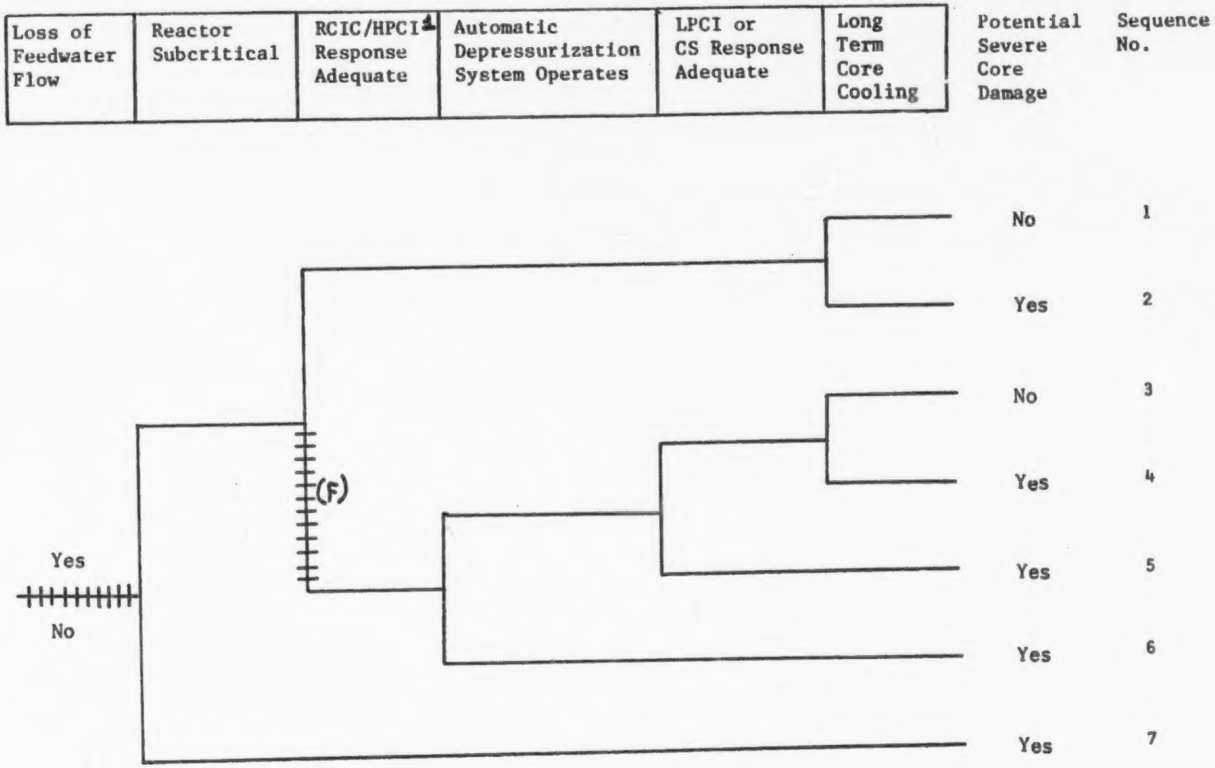
Unavailability of component per WASH 1400:* -

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor trip during power accession due to condensate system trip
HPCI failed to initiate on low-low reactor water inventory due to failed turbine stop valve
RCIC system was out of service
Reactor water level controlled by condensate system which had been immediately repaired
Potential Severe Core Damage
No
No - ADS/LPCI & CS available
No
No

NSIC 149961 — Actual Occurrence for HPCI Fails to Start When Its Turbine Stop Valve Fails to Open at Hatch 2


Loss of Feedwater Flow
Reactor Subcritical
RCIC/HPCI[1] Response Adequate
Automatic Depressurization System Operates
LPCI or CS Response Adequate
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
(F)
No 1
Yes 2
No 3
Yes 4
Yes 5
Yes 6
Yes 7

NSIC 149961 — Sequence of Interest for HPCI Fails to Start When the Turbine Stop Valve Fails to Open at Hatch 2

[1] RCIC was out of service.

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 149961

DATE OF LER: June 11, 1979

DATE OF EVENT: June 3, 1979

SYSTEM INVOLVED: HPCI

COMPONENT INVOLVED: water line, valves

CAUSE: valve improperly closed, leaky water line, (human error)

SEQUENCE OF INTEREST: loss of feedwater flow

ACTUAL OCCURRENCE: HPCI fails to start when turbine stop valve fails to open at Hatch 2

REACTOR NAME: Hatch 2

DOCKET NUMBER: 50-366

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 784 MWe

REACTOR AGE: .9 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: Southern Services/Bechtel

OPERATORS: Georgia Power Company

LOCATION: 11 miles N of Baxley, Ga.

DURATION: N/A

PLANT OPERATING CONDITION: hot shutdown

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start;
[✓ (c) made inoperable]; (d) ____________________

DISCOVERY METHOD: operational event

COMMENT: -

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 149962

Date: May 5, 1979

Title: Diesel Generator Unable to Carry Rated Load at Hatch 1

The failure sequence was:

1. Diesel generator IC was inoperable due to 5 year preventative maintenance work.
2. Diesel generator 1A was unable to carry 50% of its rated load during operability test. This was caused by improper realignment of the synchronizer gear after a modification.

Corrective action:

The synchronizer gear was properly aligned and the diesel was satisfactorily tested.

Design purpose of failed system or component:

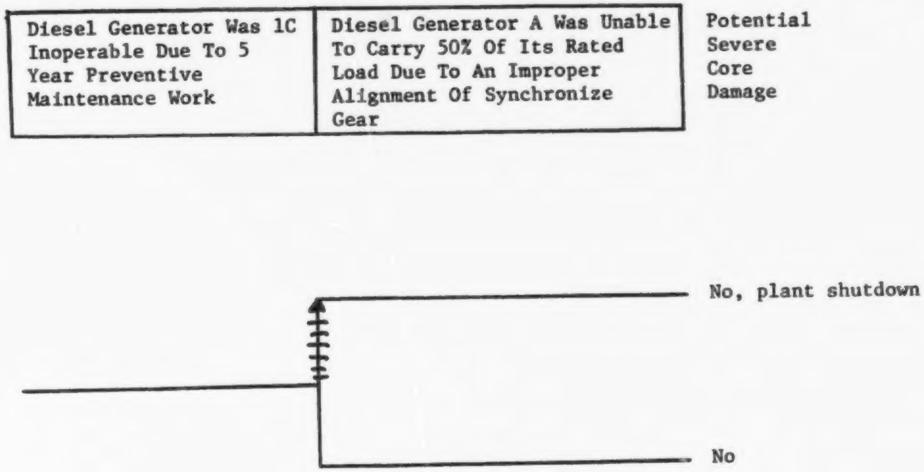
The diesel generator provides a.c. power when offsite sources are unavailable.

Unavailability of system per WASH 1400:* Emergency Power: 1×10^{-3}/D

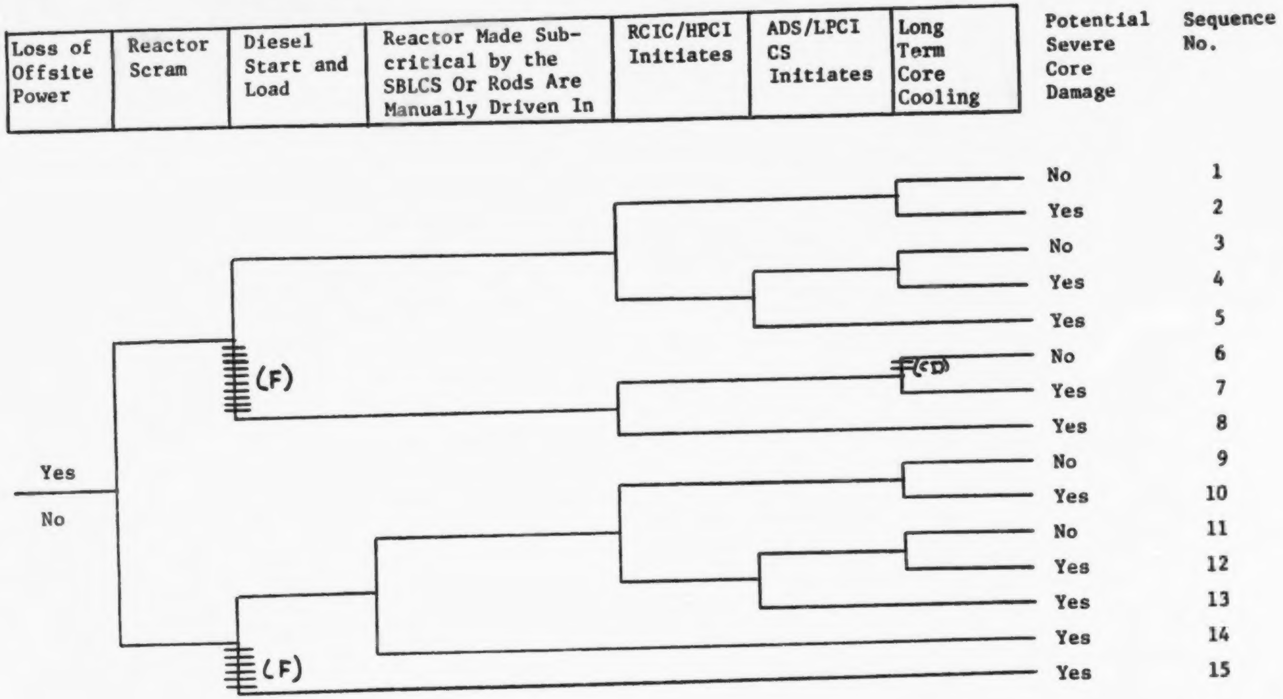
Unavailability of component per WASH 1400:* Diesel Generator failure to run: 3×10^{-4}/Hr.

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Diesel Generator Was 1C Inoperable Due To 5 Year Preventive Maintenance Work
Diesel Generator A Was Unable To Carry 50% Of Its Rated Load Due To An Improper Alignment Of Synchronize Gear
Potential Severe Core Damage
No, plant shutdown
No

NSIC 149962 — Actual Occurrence of a Diesel Generator Unable to Carry Rated Load at Hatch 1


Loss of Offsite Power
Reactor Scram
Diesel Start and Load
Reactor Made Subcritical by the SBLCS Or Rods Are Manually Driven In
RCIC/HPCI Initiates
ADS/LPCI CS Initiates
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
(F)
(F)
(F)
No 1
Yes 2
No 3
Yes 4
Yes 5
No 6
Yes 7
Yes 8
No 9
Yes 10
No 11
Yes 12
Yes 13
Yes 14
Yes 15

NSIC 149962 — Sequence of Interest for Diesel Generator Unable to Carry Load at Hatch 1

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 149962

DATE OF LER: June 7, 1979

DATE OF EVENT: May 5, 1979

SYSTEM INVOLVED: Electric Power

COMPONENT INVOLVED: Diesel Generator

CAUSE: Human Maintenance Error

SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE: Diesel Generator unable to carry load

REACTOR NAME: Hatch 1

DOCKET NUMBER: 50-321

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 786 MWe

REACTOR AGE: 4.6 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: Southern Services

OPERATORS: Georgia Power

LOCATION: 11 miles N of Baxley, GA

DURATION: 1 (a) hours

PLANT OPERATING CONDITION: 0%

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; (c) made inoperable; [✓ (d)] degraded operation

DISCOVERY METHOD: test

COMMENT:

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 150051

Date: May 30, 1979

Title: Failure of Two Diesel Generators at Dresden 2

The failure sequence was:

1. The unit 2 diesel generator had failed for reasons unspecified.
2. An operability test of unit 2/3 was immediately performed, however, it failed to start.
3. The reactor was placed in cold shutdown.

Corrective action:

1. The starting logic was modified.

Design purpose of failed system or component:

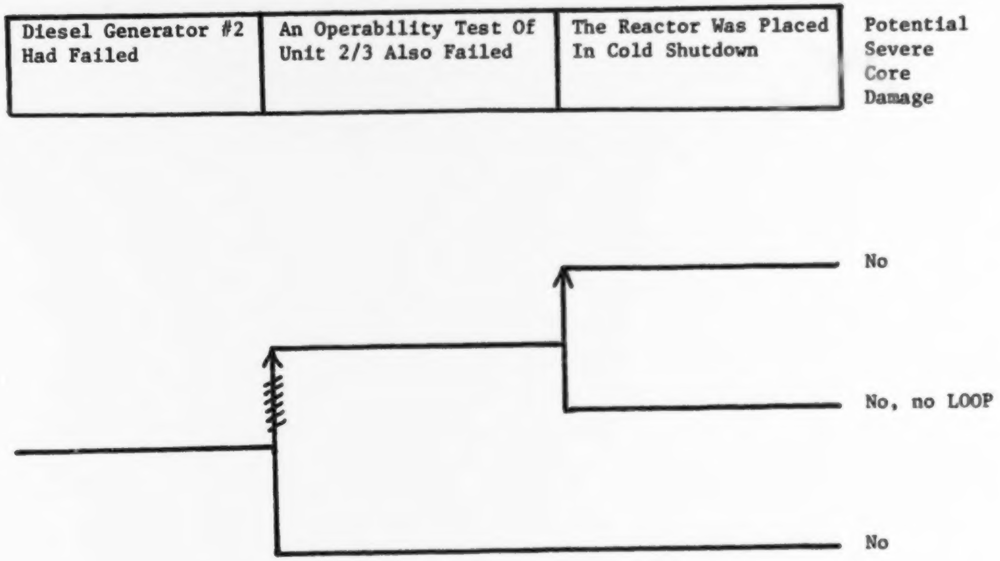
Emergency Power provides a.c. power when offsite power is unavailable.

Unavailability of system per WASH 1400:* Emergency Power: 1×10^{-3}/D

Unavailability of component per WASH 1400:* Diesel Generator Failure to Start: 3×10^{-3}/D

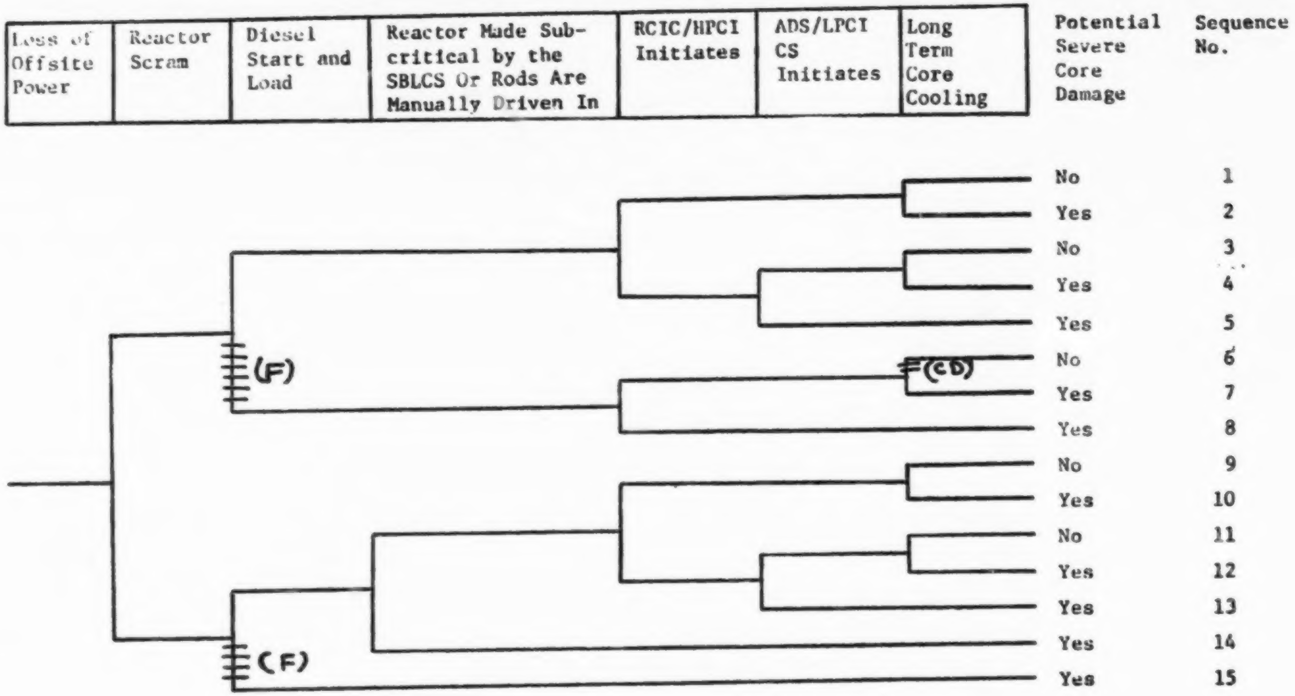
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Diesel Generator #2 Had Failed
An Operability Test Of Unit 2/3 Also Failed
The Reactor Was Placed In Cold Shutdown
Potential Severe Core Damage
No
No, no LOOP
No

NSIC 150051 — Actual Occurrence of Failure of Two Diesels at Dresden 2

625


Loss of Offsite Power
Reactor Scram
Diesel Start and Load
Reactor Made Subcritical by the SBLCS Or Rods Are Manually Driven In
RCIC/HPCI Initiates
ADS/LPCI CS Initiates
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
(F)
(CD)
(F)
No 1
Yes 2
No 3
Yes 4
Yes 5
No 6
Yes 7
Yes 8
No 9
Yes 10
No 11
Yes 12
Yes 13
Yes 14
Yes 15

NSIC 150051 — Sequence of Interest for Failure of Two Diesel Generators at Dresden 2

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 150051

DATE OF LER: June 12, 1979

DATE OF EVENT: May 30, 1979

SYSTEM INVOLVED: Emergency Power

COMPONENT INVOLVED: Diesel Generators

CAUSE: Failure of the air starter motor

SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE: Failure of two diesels

REACTOR NAME: Dresden 2

DOCKET NUMBER: 50-237

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 794 MWe

REACTOR AGE: 9.4 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: Sargent and Lundy

OPERATORS: Commonwealth Edison

LOCATION: Nine miles East of Morris, Ill.

DURATION: 360(a) hours

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performace; (b) failed to start; [✓ (c)] made inoperable; (d) ____________________

DISCOVERY METHOD: Test

COMMENT:

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 150271

Date: June 19, 1979

Title: Both RHR Service Water Pumps Inoperable at Brunswick 2

The failure sequence was:

1. The reactor was operating at 88% power.
2. The operator gave a mechanic clearance to uncouple 2B RHR pump for alignment check.
3. Mechanic uncoupled pump 2A by mistake.

Corrective action:

Seven hours later the error was discovered and pump 2A was recoupled and tested to ensure operability.

Design purpose of failed system or component:

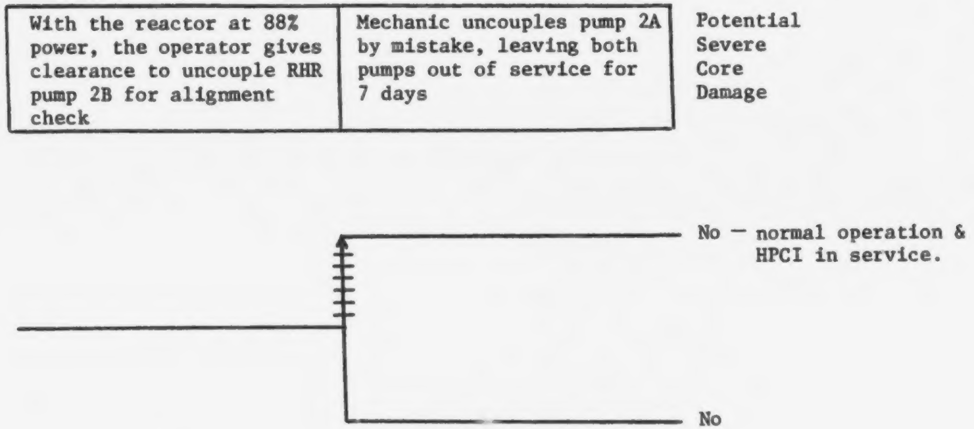
The RHR provides reactor core cooling and water level control during normal and scrammed shutdown periods. The RHR can also be operated in the LPCI mode.

Unavailability of system per WASH 1400:* 1.1×10^{-4}/D

Unavailability of component per WASH 1400:* 3×10^{-3}/D operator error

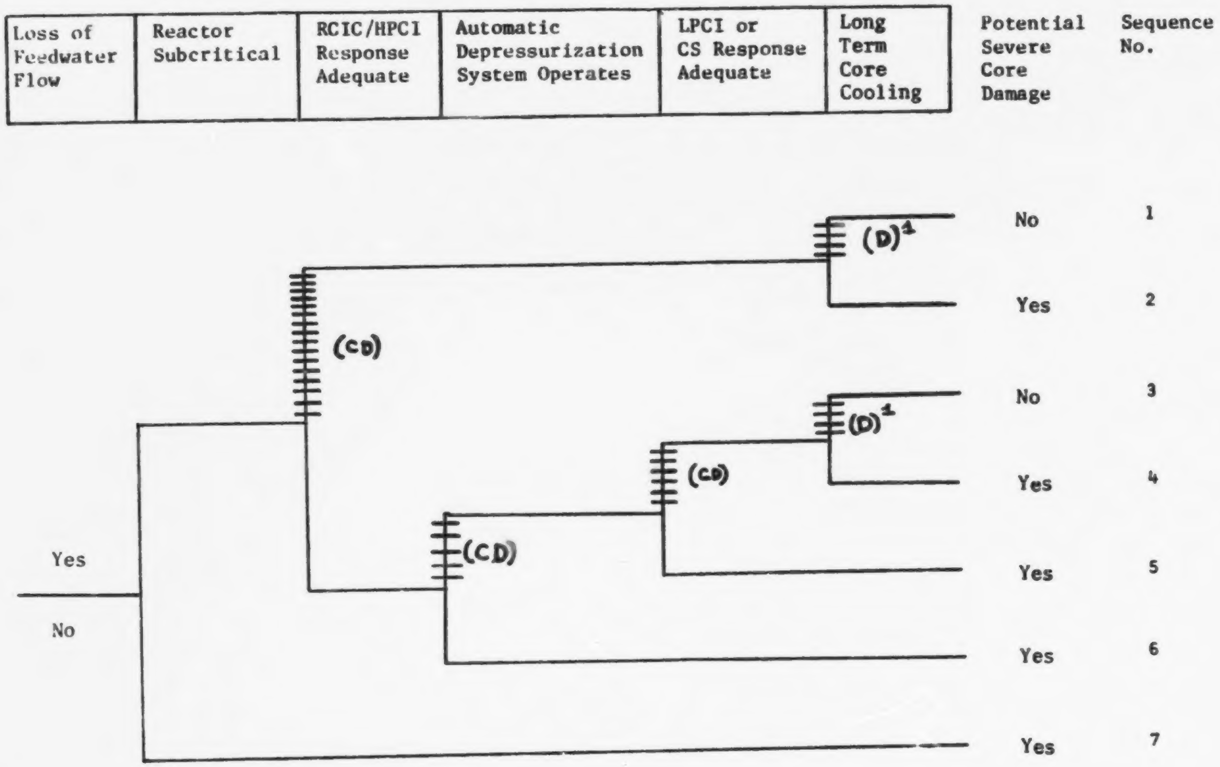
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


With the reactor at 88% power, the operator gives clearance to uncouple RHR pump 2B for alignment check
Mechanic uncouples pump 2A by mistake, leaving both pumps out of service for 7 days
Potential Severe Core Damage
No — normal operation & HPCI in service.
No

NSIC 150271 — Actual Occurrence of Both RHR Service Water Pumps Inoperable at Brunswick 2

BEST DOCUMENT AVAILABLE



Loss of Feedwater Flow
Reactor Subcritical
RCIC/HPCI Response Adequate
Automatic Depressurization System Operates
LPCI or CS Response Adequate
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
(CD)
(CD)
(CD)
(D)¹
(D)¹
No
Yes
No
Yes
Yes
Yes
Yes
1
2
3
4
5
6
7

NSIC 150271 — Sequence of Interest of Both RHR Service Water Pumps Inoperable at Brunswick 2

[1] For success the operator must rack the pumps backin within 2-24 hours.

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 150271

DATE OF LER: July 12, 1979

DATE OF EVENT: June 19, 1979

SYSTEM INVOLVED: RHR

COMPONENT INVOLVED: service water pumps

CAUSE: human error

SEQUENCE OF INTEREST: loss of feedwater

ACTUAL OCCURRENCE: Both RHR service water pumps inoperable at Brunswick 2

REACTOR NAME: Brunswick 2

DOCKET NUMBER: 50-324

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 821 MWe

REACTOR AGE: 4.25 yr

VENDOR: GE

ARCHITECT-ENGINEERS: United Engineers and Constructors

OPERATORS: Carolina Power and Light

LOCATION: Three miles N of South Port, NC

DURATION: 7 hours

PLANT OPERATING CONDITION: 88% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c) made inoperable]; (d) ____________

DISCOVERY METHOD: operator surveillance

COMMENT: —

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: **150499**

Date: **July 28, 1979**

Title: **RCIC and HPCI Isolate Following a Reactor Scram at Hatch 2**

The failure sequence was:

1. **During a MSIV fast closure test, the reactor tripped (as intended) due to MSIV closure.**
2. **HPCI and RCIC isolated due to a high ΔP signal.**
3. **The operators reset and manually started both systems.**

Corrective action:

1. **One ΔP switch was out of specs and it was reset.**
2. **An investigation was initiated to discern the cause of the out-of-spec condition.**

Design purpose of failed system or component:

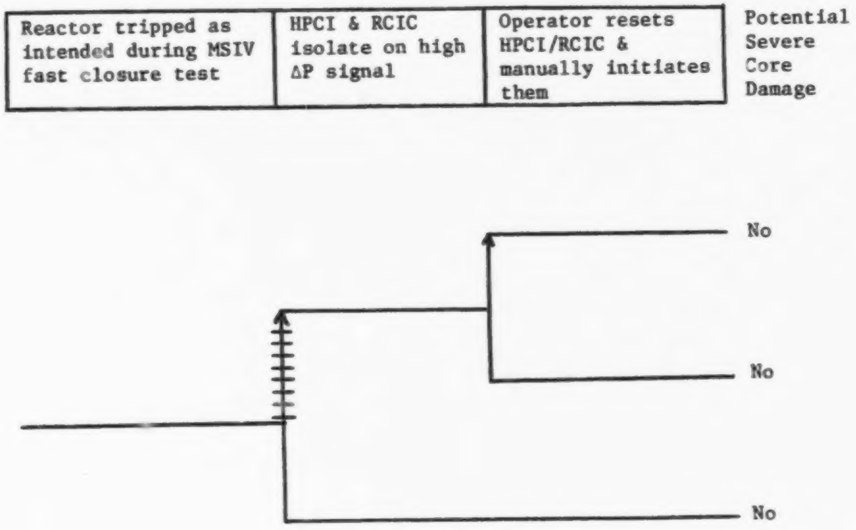
1. **HPCI is designed to maintain water level control given a small Loca.**
2. **RCIC is designed to maintain water level control given a loss of feedwater flow.**
3. **The ΔP switches are designed to send an isolation signal given a steamline pipe break in the HPCI or RCIC system.**

Unavailability of system per WASH 1400:* **HPCI: 8.8 × 10^{-2}/D given Loca**
RCIC: 8.0 × 10^{-2}/D given Loca

Unavailability of component per WASH 1400:* **pressure switches: 10^{-4}/D**

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

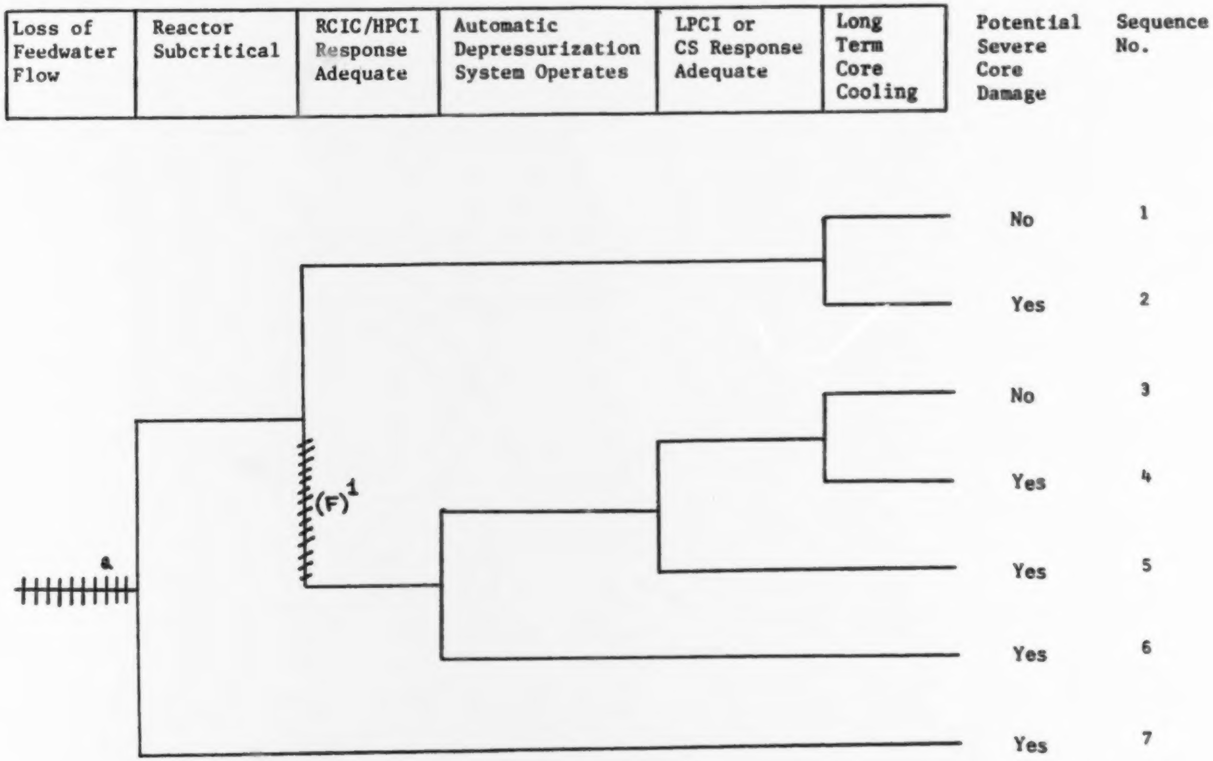

Reactor tripped as intended during MSIV fast closure test
HPCI & RCIC isolate on high ΔP signal
Operator resets HPCI/RCIC & manually initiates them
Potential Severe Core Damage
No
No
No

- Feedwater system, available as was the ADS, LPCI/CS system

NSIC 150499 — Actual Occurrence of RCIC and HPCI Isolate Following a Reactor Scram at Hatch 2

BEST DOCUMENT AVAILABLE

BEST DOCUMENT AVAILABLE


Loss of Feedwater Flow
Reactor Subcritical
RCIC/HPCI Response Adequate
Automatic Depressurization System Operates
LPCI or CS Response Adequate
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
a
(F)1
No
1
Yes
2
No
3
Yes
4
Yes
5
Yes
6
Yes
7

NSIC 150499 — Sequence of Interest for RCIC and HPCI Isolate Following a Reactor Scram at Hatch 2

[1] Success requires the operator to reset HPCI and/or RCIC.

[2] Loss of feedwater is presumed.

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 150499

DATE OF LER: July 6, 1979

DATE OF EVENT: June 28, 1979

SYSTEM INVOLVED: HPCI, RCIC

COMPONENT INVOLVED: ΔP switches

CAUSE: unknown as of reporting date

SEQUENCE OF INTEREST: loss of feedwater flow

ACTUAL OCCURRENCE: RCIC and HPCI isolate following a reactor scram at Hatch 2

REACTOR NAME: Hatch 2

DOCKET NUMBER: 50-366

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 784 MWe

REACTOR AGE: 1.0 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: Southern Services/Bechtel

OPERATORS: Georgia Power

LOCATION: 11 miles N of Baxley, Georgia

DURATION: 360(a) hours

PLANT OPERATING CONDITION: 0% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________________

DISCOVERY METHOD: operational test

COMMENT: -

635

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 150742

Date: June 22, 1979

Title: Main Stock Sampling System Lost at Peach Bottom 2

The failure sequence was:

1. During a severe electrical storm lightening struck at or near the Unit 2 main stack.
2. This resulted in initiation of HPCI and RCIC on Unit 3, tripped the unit 3 startup and emergency feed with emergency transfer to unit 2, and initiated a half scram in both units. The ability to autostart HPCI and RCIC was lost on unit 3 due to the inverter supplying power to the HPCI and RCIC.
3. A shutdown of unit 3 was commenced due to the inability of HPCI and RCIC to start automatically.

Corrective action:

1. The blown fuses in the inverter were replaced.

Design purpose of failed system or component:

1. RCIC provides a source of water during loss of feedwater event.
2. HPCI provides a source of high pressure water during a small LOCA.

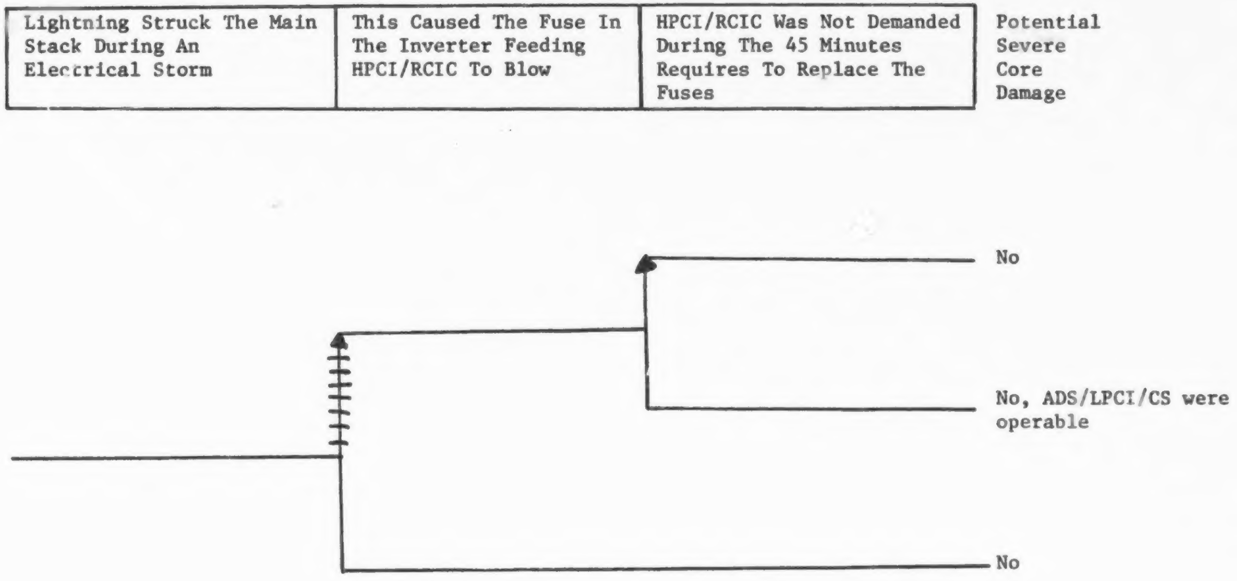
Unavailability of system per WASH 1400:* HPCI: 8.8×10^{-2}/D

RCIC: 8.0×10^{-2}/D

Unavailability of component per WASH 1400:* FUSES, Premature open: 1.10^{-6}/Hr.

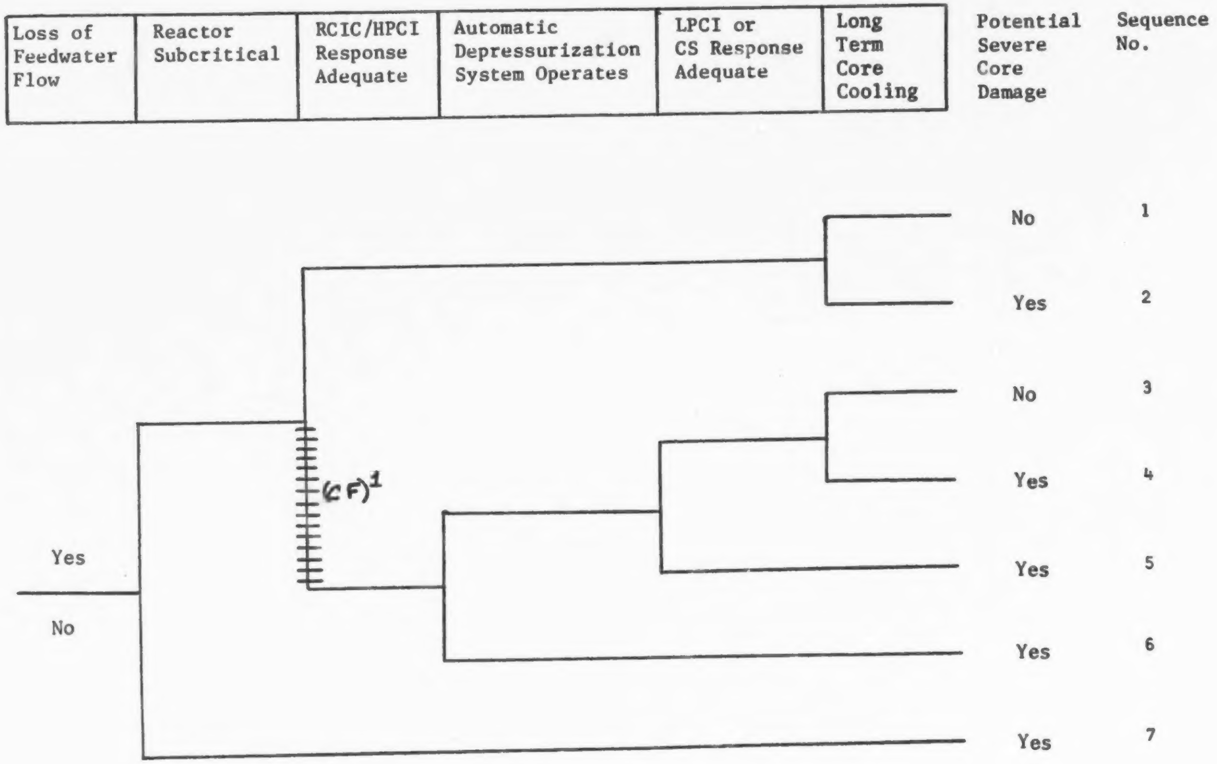
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Lightning Struck The Main Stack During An Electrical Storm
This Caused The Fuse In The Inverter Feeding HPCI/RCIC To Blow
HPCI/RCIC Was Not Demanded During The 45 Minutes Requires To Replace The Fuses
Potential Severe Core Damage
No
No, ADS/LPCI/CS were operable
No

NSIC 150742 — Actual Occurrence of Main Stack Sampling System Lost at Peach Bottom 2

637


Loss of Feedwater Flow
Reactor Subcritical
RCIC/HPCI Response Adequate
Automatic Depressurization System Operates
LPCI or CS Response Adequate
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
(CF)¹
No 1
Yes 2
No 3
Yes 4
Yes 5
Yes 6
Yes 7

NSIC 150742 — Sequence of Interest of Main Stack Sampling System Lost at Peach Bottom 2

[1] RCIC and HPCI were probably operable through a manual bus transfer and manual initiation.

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 150742

DATE OF LER: July 20, 1979

DATE OF EVENT: June 22, 1979

SYSTEM INVOLVED: HPCI/RCIC

COMPONENT INVOLVED: Inverter fuse

CAUSE: Lightning striking at or near the mainstack during an electrical storm.

SEQUENCE OF INTEREST: Loss of Feedwater

ACTUAL OCCURRENCE: Main Stack Sampling System Lost

REACTOR NAME: Peach Bottom 3

DOCKET NUMBER: 278

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 1065 MWe

REACTOR AGE: 5.9 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS:

OPERATORS: Philadelphia Electric Company

LOCATION: Nineteen miles S of Lancaster

DURATION: 5.0 hours

PLANT OPERATING CONDITION: 100% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ______________________

DISCOVERY METHOD: Operational Event

COMMENT: RCIC & HPCI are fed from the same inverter

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 150882

Date: June 6, 1979

Title: Both Diesel Generators Trip During Testing at Crystal River 3

The failure sequence was:

1. During performance of Engineered Safeguards Time Response Test, both diesel generators tripped when loaded on the emergency buses due to an imbalance in reactive load.

Corrective action:

1. A procedure revision was implemented to eliminate the high MVAR and reduce the reactive load imbalance.
2. The diesel generators were surveilled daily from June 9 — June 15 and performed satisfactorily.

Design purpose of failed system or component:

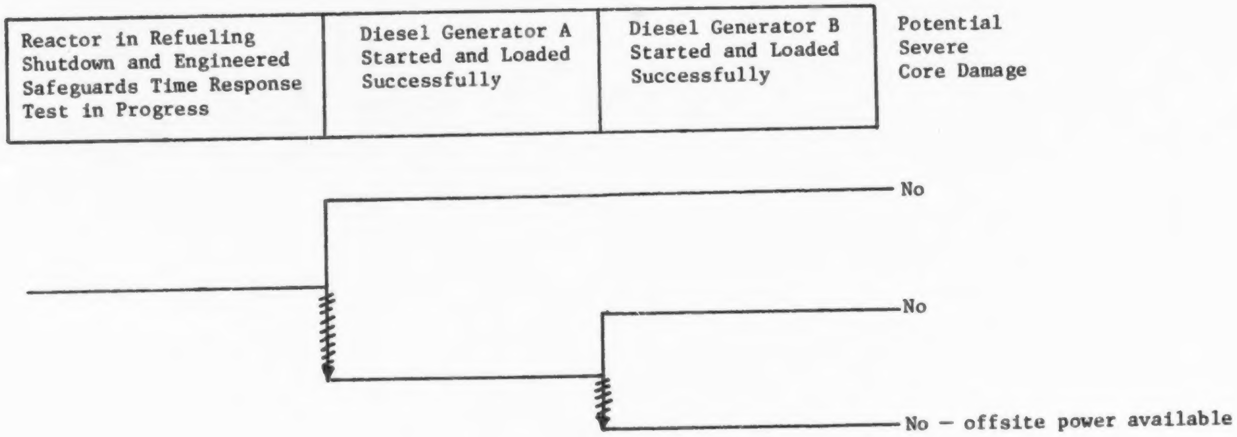
1. The emergency diesel-generators provide power to safety-related equipment in the event of loss of offsite power.

Unavailability of system per WASH 1400:* emergency power systems: 1×10^{-2}/D

Unavailability of component per WASH 1400:* each diesel generator: 3×10^{-2}/D

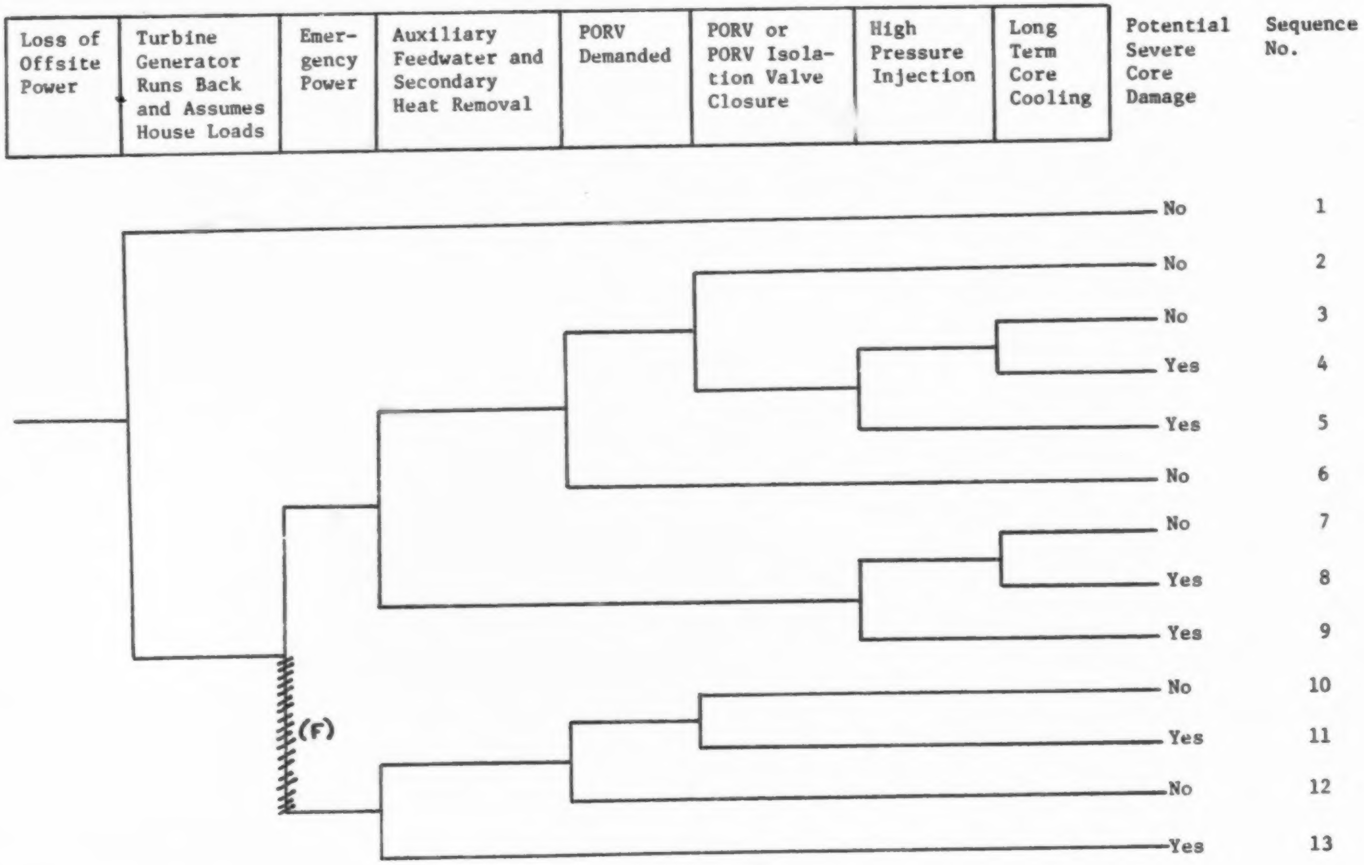
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor in Refueling Shutdown and Engineered Safeguards Time Response Test in Progress
Diesel Generator A Started and Loaded Successfully
Diesel Generator B Started and Loaded Successfully
Potential Severe Core Damage
No
No
No – offsite power available

NSIC 150882 – Actual Occurrence for Failure of Both Diesel Generators During Testing at Crystal River 3

BEST DOCUMENT AVAILABLE


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
Yes 9
No 10
Yes 11
No 12
Yes 13
(F)

NSIC 150882 — Sequence of Interest for Failure of Both Diesel Generators During Testing at Crystal River 3

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 150882

DATE OF LER: June 27, 1979

DATE OF EVENT: June 6, 1979

SYSTEM INVOLVED: Emergency Electric Power

COMPONENT INVOLVED: 2 Diesel Generators

CAUSE: Imbalance in reactive load, both DGs tripped, (human error)

SEQUENCE OF INTEREST: Loss of off-site power

ACTUAL OCCURRENCE: Both DGs tripped during test

REACTOR NAME: Crystal River 3

DOCKET NUMBER: 50-302

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 825 MWe

REACTOR AGE: 2.4 yr

VENDOR: B&W

ARCHITECT-ENGINEERS: Gilbert Associates

OPERATORS: Florida Power Corp.

LOCATION: 7 miles NW of Crystal River, Fla.

DURATION: 360(a) hours

PLANT OPERATING CONDITION: 0% power

SAFETY FEATURE TYPE OF FAILURE: [✓ (a)] inadequate performance; (b) failed to start; (c) made inoperable; (d) ____________________

DISCOVERY METHOD: Testing

COMMENT: —

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: **150943**

Date: **July 27, 1979**

Title: **Reactor Scram From Loss of Offsite Power at Pilgrim 1**

The failure sequence was:

1. **The reactor was operating at 100% when lightning struck the offsite transmission lines.**
2. **This caused backflashing which resulted in a high speed relay operation for a line to ground fault.**
3. **The reactor tripped.**
4. **The diesel generators started and loaded properly.**
5. **Relief/Safety valve 203-3D was manually actuated to control pressure.**

Corrective action:

1. **Offsite power was restored in ~1 1/2 hours.**

Design purpose of failed system or component:

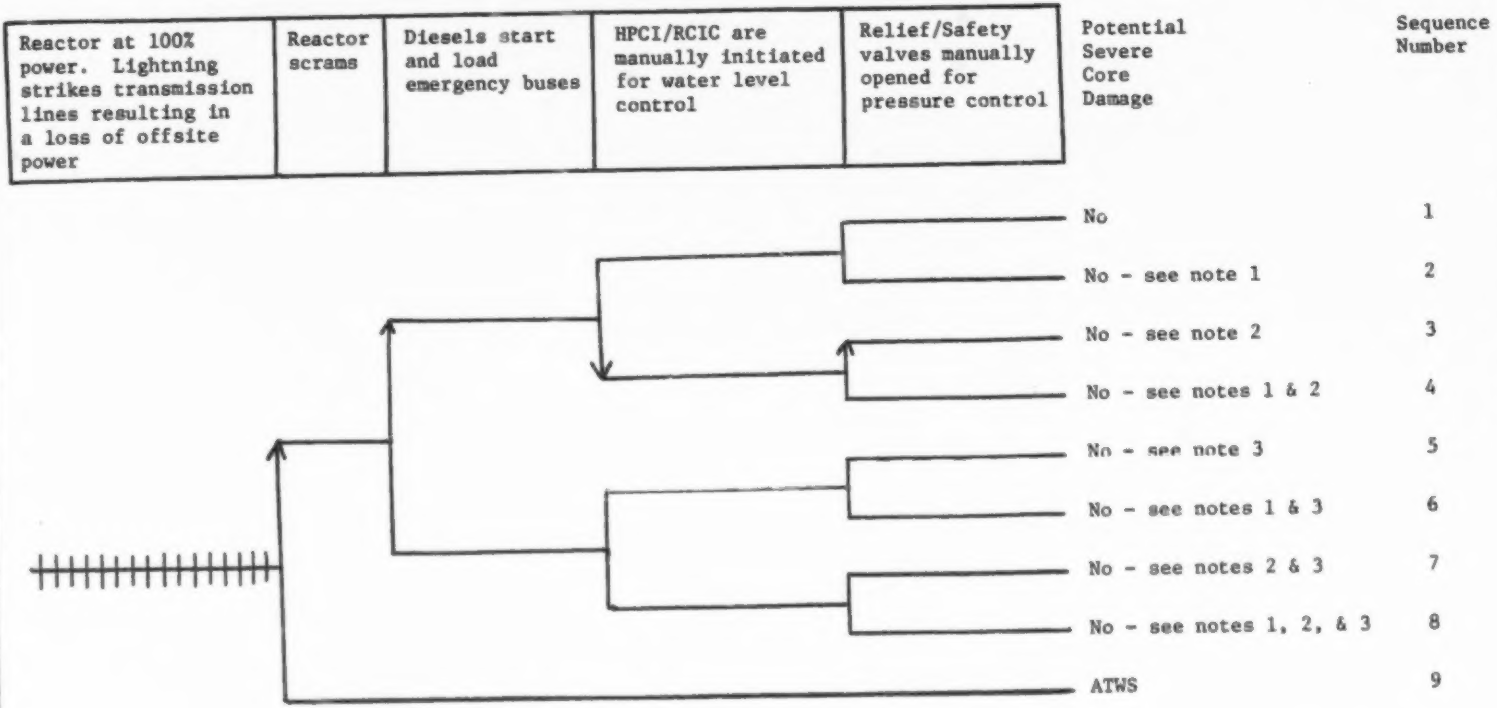
The transmission lines tie Pilgrim 1 to the New England grid. These lines are a source of power when the unit isn't generating power.

Unavailability of system per WASH 1400:* **loss of offsite power 2.0×10^{-5}/hrs**

Unavailability of component per WASH 1400:* **-**

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor at 100% power. Lightning strikes transmission lines resulting in a loss of offsite power
Reactor scrams
Diesels start and load emergency buses
HPCI/RCIC are manually initiated for water level control
Relief/Safety valves manually opened for pressure control
Potential Severe Core Damage
Sequence Number
No
1
No - see note 1
2
No - see note 2
3
No - see notes 1 & 2
4
No - see note 3
5
No - see notes 1 & 3
6
No - see notes 2 & 3
7
No - see notes 1, 2, & 3
8
ATWS
9

NSIC 150943 — Actual Occurrence of the Loss of Offsite Power at Pilgrim 1

[redacted]

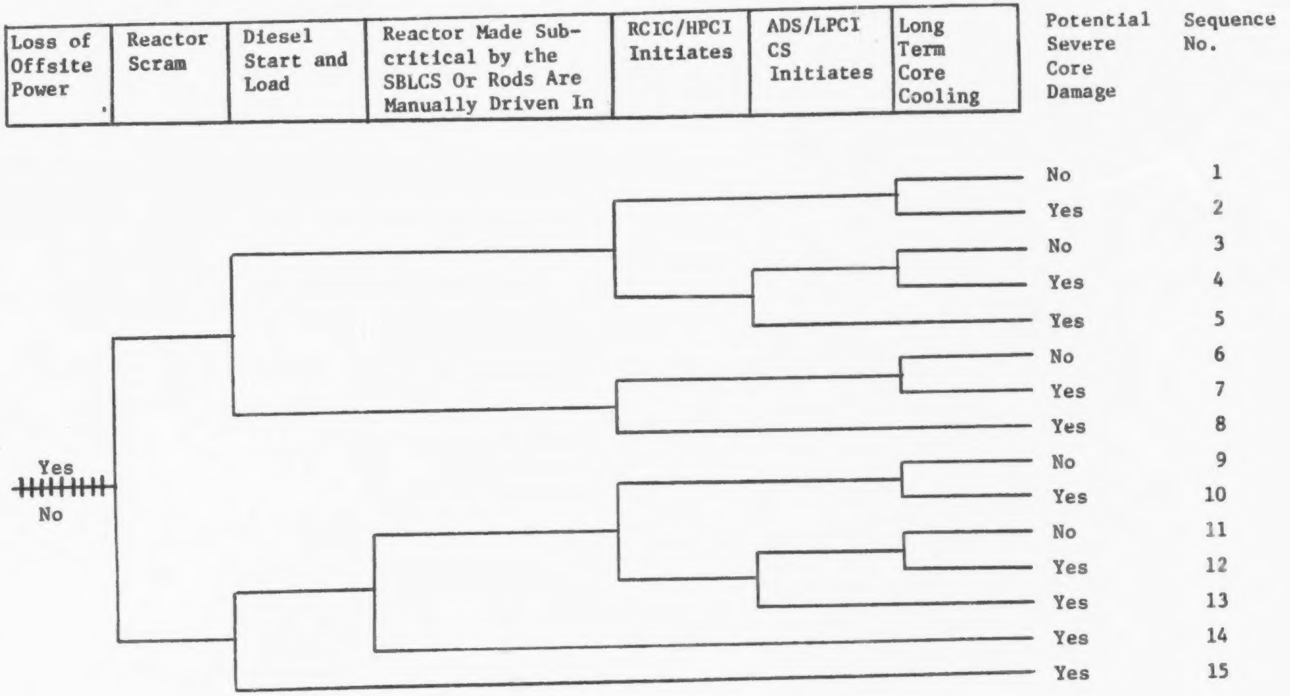
Notes

1. The safety and safety/relief valves automatically opened at the following set points:

	No. of Valves	Set Pressure (PSIG)
Relief valves	3	1090 to 1100
Safety valves	2	1240

2. The HPCI and the RCIC autoactuate at either low-low reactor water inventory or high dry well pressure. See note 1.

3. The HPCI and RCIC are capable of initiation without an a.c. power source. The safety and relief valves also have this capability. The battery system is capable of 8 hours at 77°F operation without recharge. Since offsite power was restored within ∿1½ hour, a potential core damage situation did not exist.


Loss of Offsite Power
Reactor Scram
Diesel Start and Load
Reactor Made Subcritical by the SBLCS Or Rods Are Manually Driven In
RCIC/HPCI Initiates
ADS/LPCI CS Initiates
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
No 1
Yes 2
No 3
Yes 4
Yes 5
No 6
Yes 7
Yes 8
No 9
Yes 10
No 11
Yes 12
Yes 13
Yes 14
Yes 15

NSIC 150943 — Sequence of Interest of the Loss of Offsite Power at Pilgrim 1

647

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 150943

DATE OF LER: August 10, 1979

DATE OF EVENT: July 27, 1979

SYSTEM INVOLVED: electric power

COMPONENT INVOLVED: grounding relays

CAUSE: lightning struck offsite transmission lines causing high speed relay operation for line to ground fault.

SEQUENCE OF INTEREST: loss of offsite power

ACTUAL OCCURRENCE: loss of offsite power

REACTOR NAME: Pilgrim 1

DOCKET NUMBER: 50-293

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 655 MWe

REACTOR AGE: 7.1 yr

VENDOR: GE

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Boston Edison

LOCATION: 4 miles SE of Plymouth

DURATION: N/A

PLANT OPERATING CONDITION: 100% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ______________________

DISCOVERY METHOD: operational event

COMMENT: -

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 151433

Date: August 13, 1979

Title: Incorrect RHR Service Water Pump Breakers Racked Out at Brunswick 2

The failure sequence was:

1. An attempt was made to place the "2B" loop RHR service water under clearance for maintenance.
2. The "2A" pump was mistakenly racked out. This resulted in both pumps being out of service for approximately 15 minutes.

Corrective action:

The control operator indicated on the tag and tag out sheet that pump "2A" was designated to be racked out instead of "2B". An auxiliary operator realized the tag sheet did not reflect the actual equipment cleared. The "2A" RHR SW pump was immediately racked back in.

Design purpose of failed system or component:

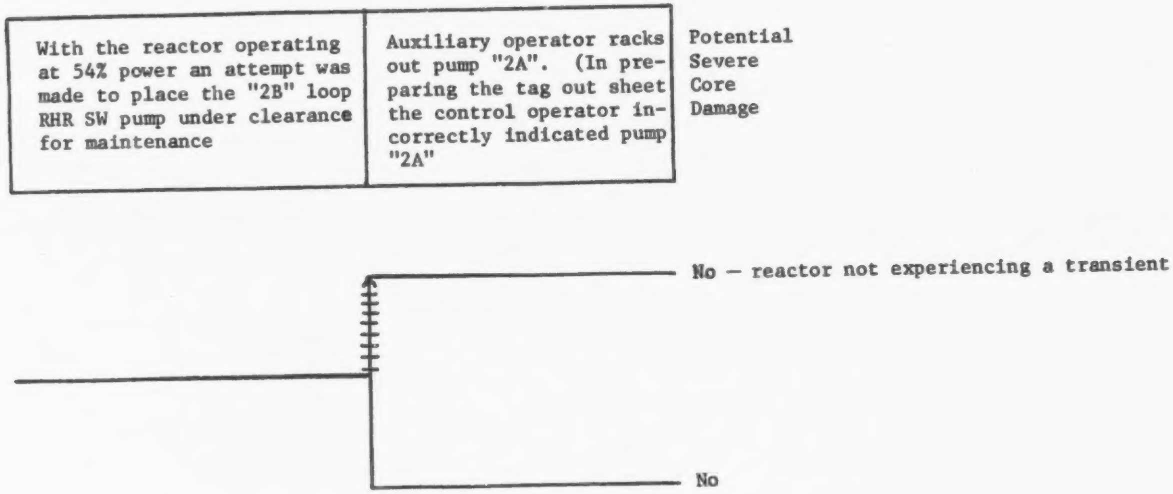
The RHR service water pump supply water to the RHR heat exchanger and thus provides a heat sink.

Unavailability of system per WASH 1400:* RHR: 1.1×10^{-4}/D

Unavailability of component per WASH 1400:* operator error: 3×10^{-3}/D

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.



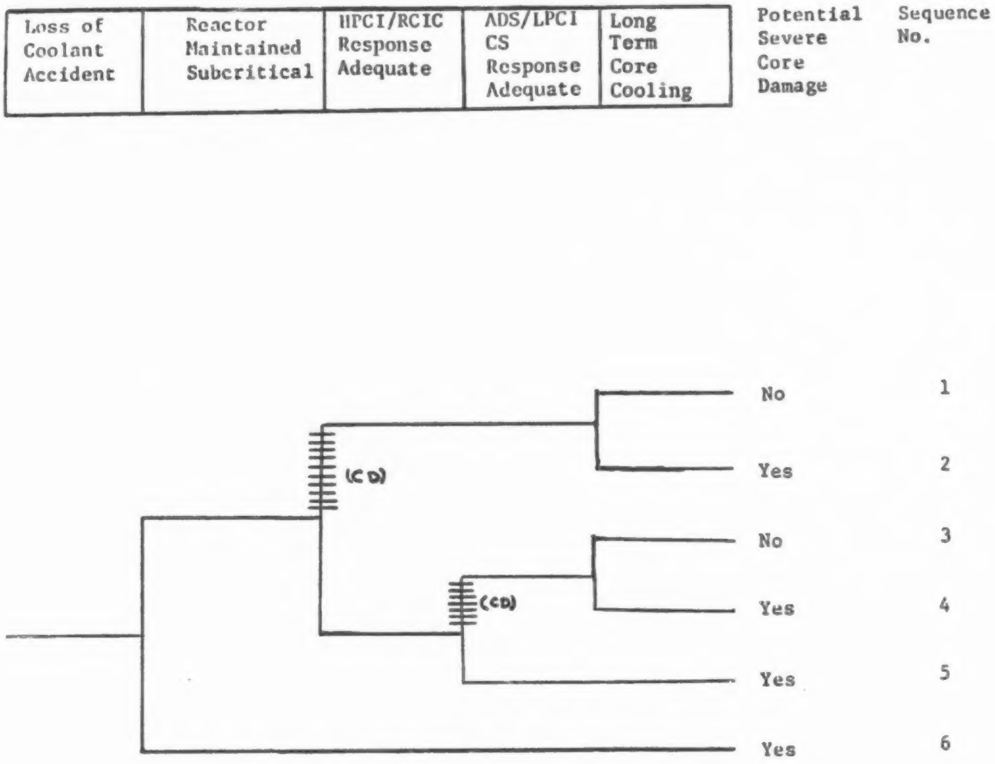
With the reactor operating at 54% power an attempt was made to place the "2B" loop RHR SW pump under clearance for maintenance

Auxiliary operator racks out pump "2A". (In preparing the tag out sheet the control operator incorrectly indicated pump "2A"

Potential
Severe
Core
Damage
No – reactor not experiencing a transient
No

NSIC 151433 – Actual Occurrence for Incorrect RHR Service Water Pump Breaker Racked Out at Brunswick 2

[redacted]


Loss of Coolant Accident
Reactor Maintained Subcritical
HPCI/RCIC Response Adequate
ADS/LPCI CS Response Adequate
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
(CD)
(CD)
No
1
Yes
2
No
3
Yes
4
Yes
5
Yes
6

NSIC 151433 — Sequence of Interest for Incorrect RHR Service Water Breakers Racked Out at Brunswick 2

[1] For success the operator must rack the pumps back in within 2 to 24 hours.

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 151433

DATE OF LER: August 13, 1979

DATE OF EVENT: September 11, 1979

SYSTEM INVOLVED: Residual Heat Removal System

COMPONENT INVOLVED: RHR service water pumps

CAUSE: Human error

SEQUENCE OF INTEREST: Loss of Coolant Accident

ACTUAL OCCURRENCE: Incorrect RHR service water pump breaker racked out at Brunswick 2

REACTOR NAME: Brunswick 2

DOCKET NUMBER: 50-324

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 821 MWe

REACTOR AGE: 4.4 yr

VENDOR: GE

ARCHITECT-ENGINEERS: United Engineers and Constructors

OPERATORS: Carolina Power and Light

LOCATION: 3 miles N of Southport

DURATION: 0.25 hours

PLANT OPERATING CONDITION: 54% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________________

DISCOVERY METHOD: operator surveillance

COMMENT: -

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 151635

Date: **August 18, 1979**

Title: **All Offsite Power Lost at Pilgrim 1**

The failure sequence was:

1. Pilgrim 1 was operating at 100% power when lightning struck the switchyard causing a ground fault. This resulted in a total loss of offsite power.
2. The reactor tripped due to the load rejection.
3. The diesels started and loaded the vital buses.
4. A relief valve was manually opened for pressure control.
5. The reactor was brought to a safe shutdown.

Corrective action:

1. Offsite power was returned to the station in $\sim$1/2 hour.

Design purpose of failed system or component:

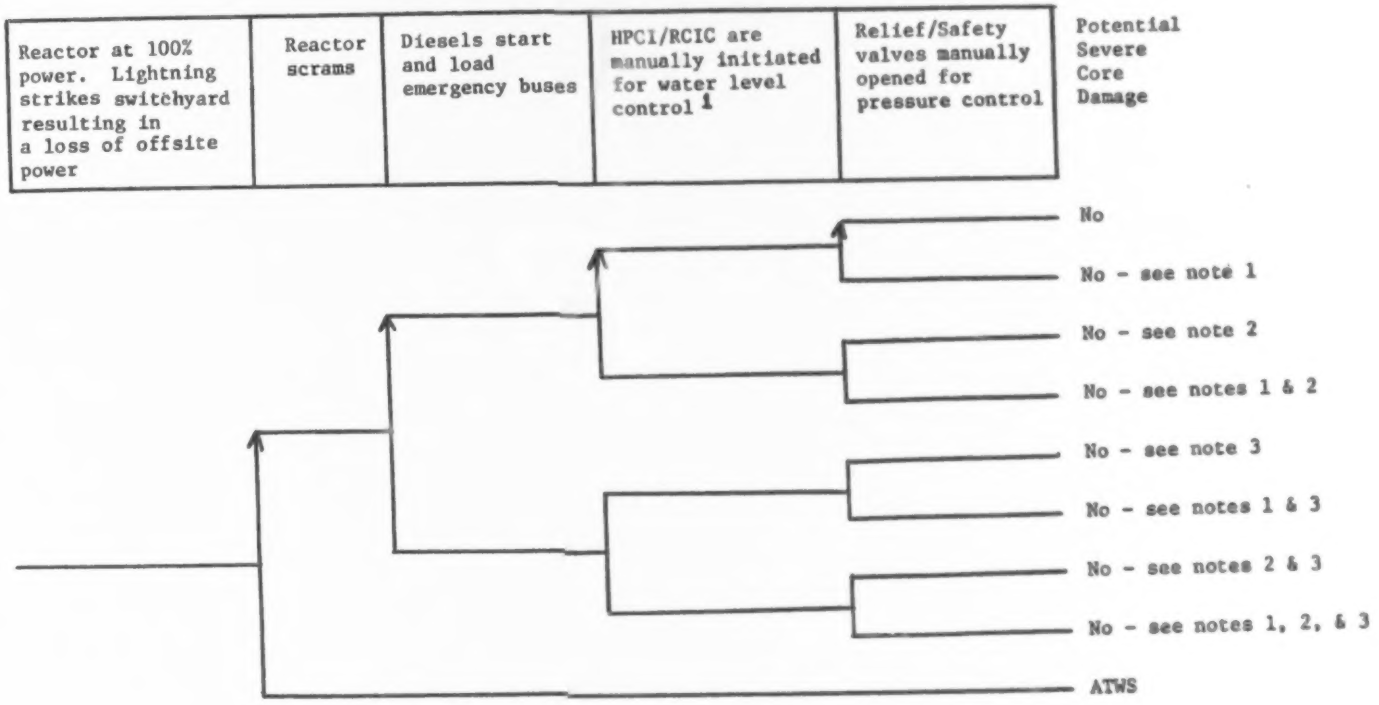
Offsite power is the source of electricity to both safety and non-safety related loads when the reactor is shutdown.

Unavailability of system per WASH 1400:* loss of offsite power: 2×10^{-5}/hr

Unavailability of component per WASH 1400:*

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor at 100% power. Lightning strikes switchyard resulting in a loss of offsite power
Reactor scrams
Diesels start and load emergency buses
HPCI/RCIC are manually initiated for water level control[1]
Relief/Safety valves manually opened for pressure control
Potential Severe Core Damage
No
No - see note 1
No - see note 2
No - see notes 1 & 2
No - see note 3
No - see notes 1 & 3
No - see notes 2 & 3
No - see notes 1, 2, & 3
ATWS

NSIC 151635 — Actual Occurrence of the Loss of Offsite Power at Pilgrim 1

[1] This action is presumed.

[redacted]

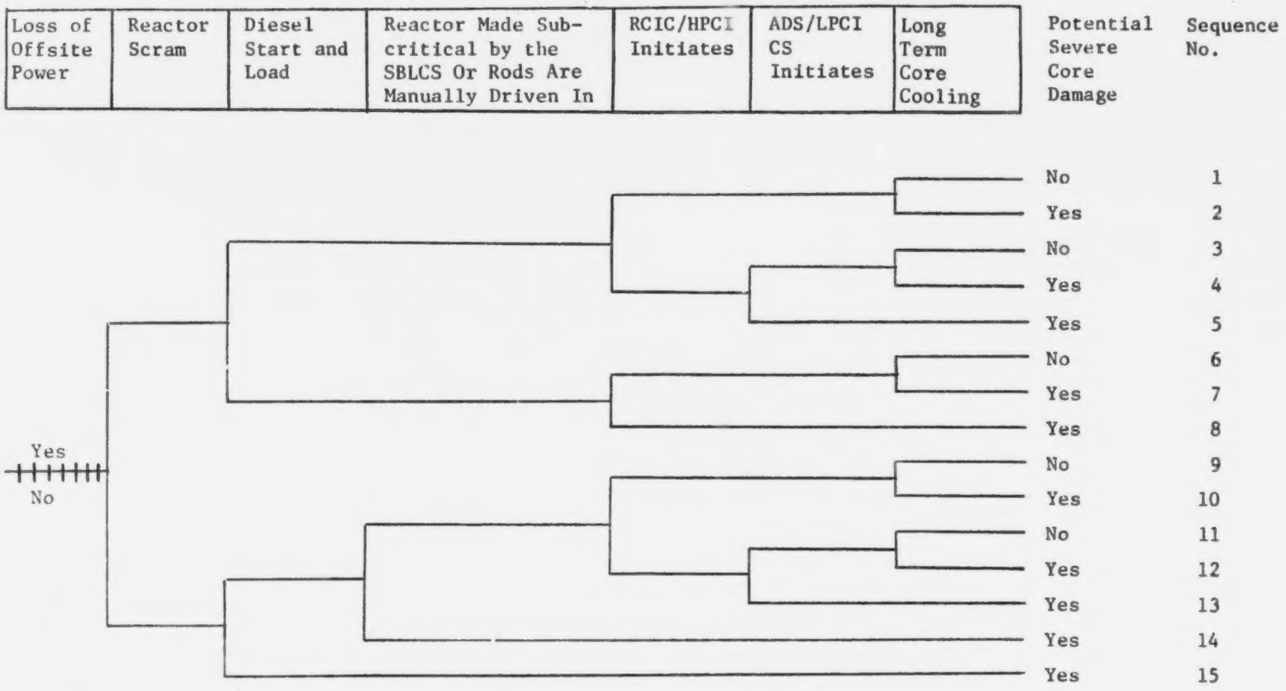
Notes

1. The safety and safety/relief valves automatically open at the following set points:

	No. of Valves	Set Pressure (PSIG)
Relief valves	3	1090 to 1240
Safety valves	2	1240

2. The HPCI and the RCIC autoactuate at either low-low reactor water inventory or high dry well pressure. See note 1.

3. The HPCI and RCIC are capable of initiation without an a.c. power source. The safety and relief valves also have this capability. The battery systems are capable of 8 hour operation without recharge. Since offsite power was restored within ~½ hour, a potential core damage situation did not exist.


Loss of Offsite Power
Reactor Scram
Diesel Start and Load
Reactor Made Sub-critical by the SBLCS Or Rods Are Manually Driven In
RCIC/HPCI Initiates
ADS/LPCI CS Initiates
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
No 1
Yes 2
No 3
Yes 4
Yes 5
No 6
Yes 7
Yes 8
No 9
Yes 10
No 11
Yes 12
Yes 13
Yes 14
Yes 15

NSIC 151635 — Sequence of Interest for the Loss of Offsite Power at Pilgrim 1

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: **151635**

DATE OF LER: **September 11, 1978**

DATE OF EVENT: **August 28, 1979**

SYSTEM INVOLVED: **electric power**

COMPONENT INVOLVED: **switchgear**

CAUSE: **lightning**

SEQUENCE OF INTEREST: **loss of offsite power**

ACTUAL OCCURRENCE: **loss of offsite power**

REACTOR NAME: **Pilgrim 1**

DOCKET NUMBER: **50-293**

REACTOR TYPE: **BWR**

DESIGN ELECTRICAL RATING: **655** MWe

REACTOR AGE: **7.2** yr

VENDOR: **GE**

ARCHITECT-ENGINEERS: **Bechtel**

OPERATORS: **Boston Edison**

LOCATION: **4 miles SE of Plymouth**

DURATION: **N/A**

PLANT OPERATING CONDITION: **100% power**

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________

DISCOVERY METHOD: **operational event**

COMMENT: **Third loop occurrence at Pilgrim.**

657

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 152183

Date: August 13, 1979

Title: Pressurizer Power Operated Relief Valve Opens at Connecticut Yankee

The failure sequence was:

1. With the reactor at 100% power, a bistable in a pressurizer pressure controller failed and resulted in the opening of pressurizer power operated relief valve V-570 and its isolation valve MOV-569 and consequent RCS blowdown.

Corrective action:

1. The operator overrode the pressure controller signal and closed the PORV isolation valve.
2. The failed bistable was repaired.

Design purpose of failed system or component:

The pressure controller provides a control signal to one of two pressurizer power operated relief valves, opening the valve when the pressurizer pressure reaches a preset point.

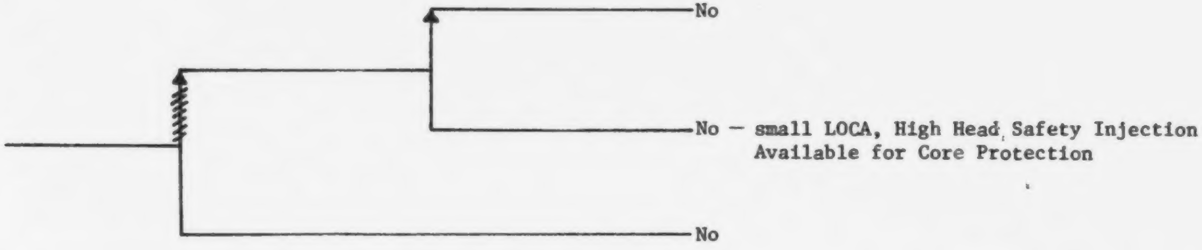
Unavailability of system per WASH 1400:* not addressed

Unavailability of component per WASH 1400:* bistable (instrumentation — low power application): 1×10^{-6}/hr.

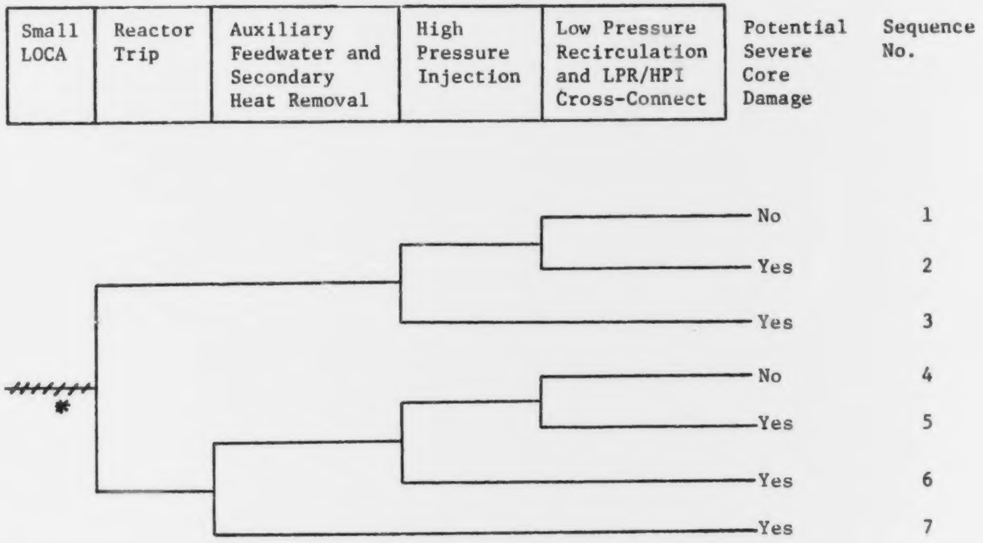
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

Reactor at 100% Power	Pressurizer Power Operated Relief Valve V-570 and Its Isolation Valve MOV-569 Open Due to Pressure Controller Bistable Failure	Operator Overrides Controller Signal and Closes PORV Isolation Valve	Potential Severe Core Damage


No
No — small LOCA, High Head Safety Injection Available for Core Protection
No

NSIC 152183 — Actual Occurrence for Pressurizer Power Operated Relief Valve Opens at Connecticut Yankee


Small LOCA
Reactor Trip
Auxiliary Feedwater and Secondary Heat Removal
High Pressure Injection
Low Pressure Recirculation and LPR/HPI Cross-Connect
Potential Severe Core Damage
Sequence No.
*
No 1
Yes 2
Yes 3
No 4
Yes 5
Yes 6
Yes 7

NSIC 152183 — Sequence of Interest for Pressurizer Power Operated Relief Valve Opens at Connecticut Yankee

*Failure requires the operator to fail to close the PORV isolation valve.

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 152183

DATE OF LER: September 7, 1979

DATE OF EVENT: August 13, 1979

SYSTEM INVOLVED: pressure relief

COMPONENT INVOLVED: power operated relief valve and its isolation valve

CAUSE: failed pressure controller bistable

SEQUENCE OF INTEREST: small LOCA

ACTUAL OCCURRENCE: open power operated relief valve and isolation valve

REACTOR NAME: Connecticut Yankee

DOCKET NUMBER: 50-213

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 575 MWe

REACTOR AGE: 12.1 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Stone & Webster

OPERATORS: Connecticut Yankee Atomic Power Co.

LOCATION: 13 miles east of Meriden, Conn.

DURATION: N/A

PLANT OPERATING CONDITION: 100% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; (c) made inoperable; [✓ (d)] failed open

DISCOVERY METHOD: operator observation

COMMENT: According to the plant's FSAR, the PORV isolation valve is not controlled by the pressure controller, only manually.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 152187

Date: September 3, 1979

Title: Startup Transformer Fails and Diesel Generator Fails to Start at St. Lucie 1

The failure sequence was:

1. With the reactor in cold shutdown during the passage of hurricane David, a support cable for a crane used in the Unit 2 construction fell across the lines to the "B" startup transformer.
2. The East switchyard bus locked-out and the "B" startup transformer de-energized, resulting in de-energization of the "B" and "AB" (because of the alignment at the time) electrical systems.
3. The "B" diesel-generator failed to start due to a binding relay in its auto-start circuitry, the "B" invertor failed due to a blown fuse, and the "AB" 125VDC bus tripped due to a current transient during the evnet, resulting in the loss of numerous instrumentation.

Corrective action:

1. The binding relay was replaced.
2. The auto-start and loading capability of the "B" diesel generator was verified.

Design purpose of failed system or component:

1. The diesel-generator provides an on-site source of electric power for safety-related loads when the unit generator and offsite sources are unavailable.
2. The invertor provides regulated 120 VAC for instrumentation loads from the plant batteries.
3. DC bus "AB" provides power to the steam inlet valves to the AFW pump turbine

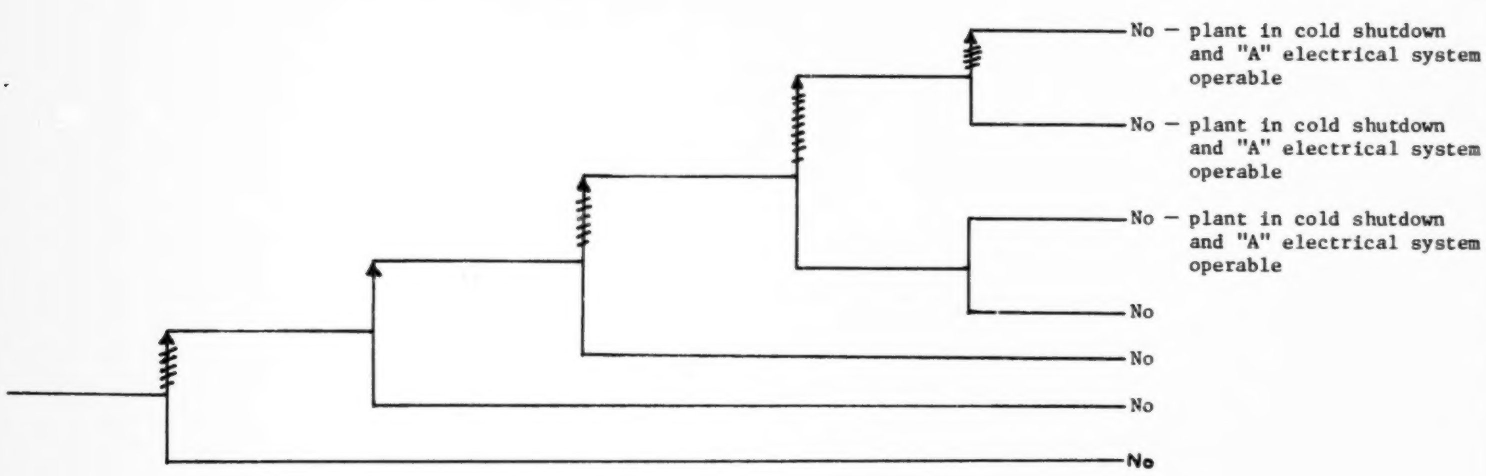
Unavailability of system per WASH 1400:* —

Unavailability of component per WASH 1400:* Diesel generator: 3×10^{-2}/D
Invertor: 3×10^{-6}/hr

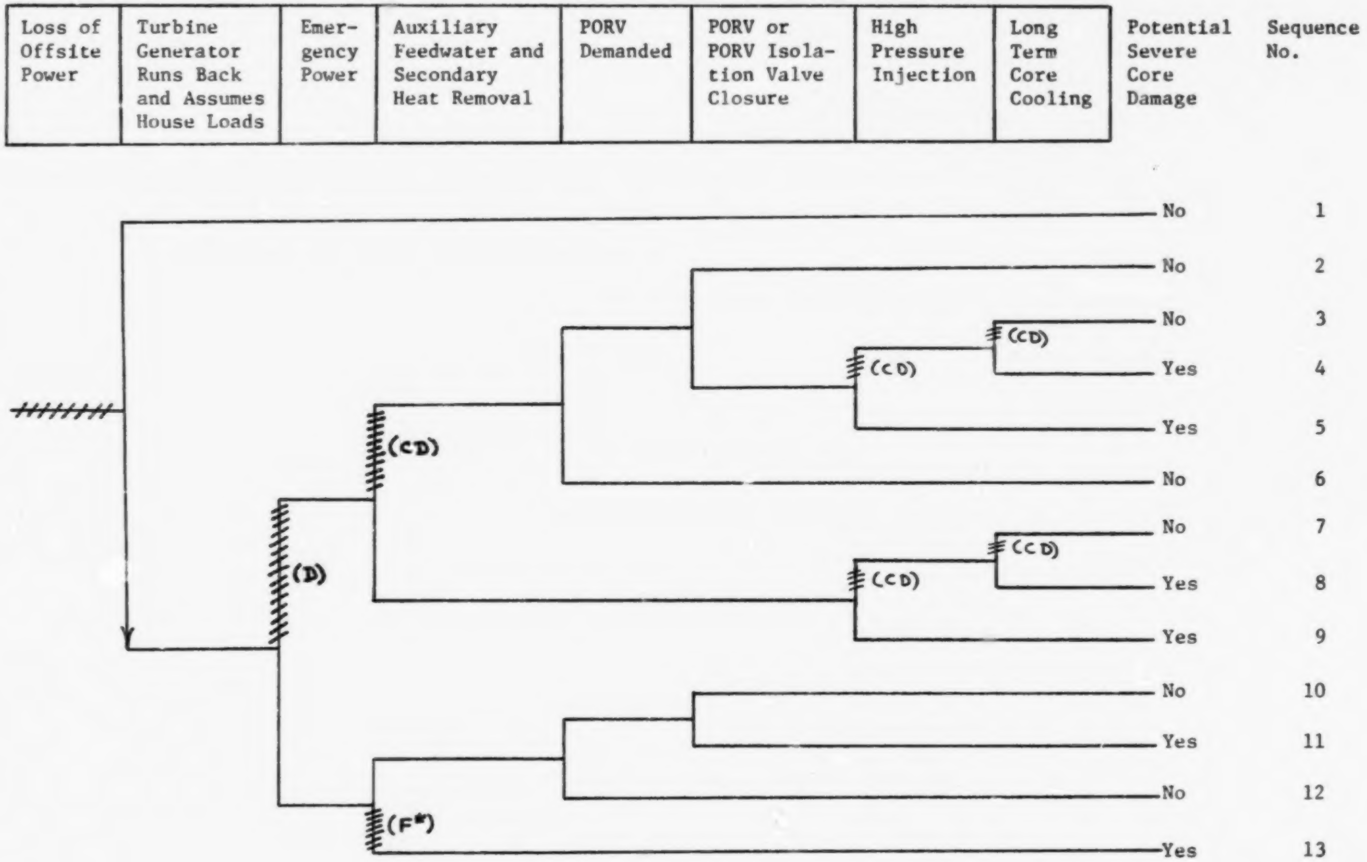
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

Reactor in Cold Shutdown During Passage of Hurricane David	Support Cable for Crane (Unit 2 Construction Falls Across Lines to "B" Startup Trans.	East Switchyard Bus Lockout and De-energization of "B" Startup Trans. ("B" and "AB" Electrical Systems De-energized)	"B" Diesel-Generator Fails to Start Due to Binding Relay in Auto-Start Circuitry	"B" Invertor Fails Due to a Blown Fuse	"AB" D.C. Bus De-energized Due to a Current Transient	Potential Severe Core Damage


No — plant in cold shutdown and "A" electrical system operable
No — plant in cold shutdown and "A" electrical system operable
No — plant in cold shutdown and "A" electrical system operable
No
No
No
No

NSIC 152187 — Actual Occurrence for Startup Transformer Fails and Diesel-Generator Fails to Start at St. Lucie 1


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emer-gency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
(D)
(CD)
(CD)
(CD)
(CD)
(CD)
(F*)
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
Yes 9
No 10
Yes 11
No 12
Yes 13

NSIC 152187 — Sequence of Interest for Startup Transformer Fails and Diesel-Generator Fails to Start at St. Lucie 1

*Manual action may be successful in initiating the turbine-driven AFW pump.

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 152187

DATE OF LER: September 17, 1979

DATE OF EVENT: September 3, 1979

SYSTEM INVOLVED: Electric power

COMPONENT INVOLVED: Diesel generator, inverter, startup transformer

CAUSE: Cable drop across startup transformer lines, binding relay in diesel generator start circuitry, blown inverter fuse.

SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE: Switchyard lockout due to cable drop during storm.

REACTOR NAME: St. Lucie 1

DOCKET NUMBER: 50-335

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 802 MWe

REACTOR AGE: 3.4 yr

VENDOR: Combustion Engineering

ARCHITECT-ENGINEERS: Ebasco

OPERATORS: Florida Power & Light Co.

LOCATION: 12 miles SE of Fort Pierce, Florida

DURATION: N/A

PLANT OPERATING CONDITION: Cold shutdown

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ______________________

DISCOVERY METHOD: Event during shutdown

COMMENT: —

665

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: **152563**

Date: **October 2, 1979**

Title: **Tube Break Occurs in Steam Generator at Prairie Island 1**

The failure sequence was:

1. **With the reactor at 100% power, a tube break occurred in the "A" steam generator.**
2. **The air ejector radiation monitor alarmed and a turbine runback occurred due to decreasing RCS pressure.**
3. **The operator commenced load reduction.**
4. **The operator started the second and third charging pumps on low pressurizer level.**
5. **Reactor Trip and Safety Injection occurred due to low pressurizer pressure.**
6. **The operator tripped both main coolant pumps.**

(Continued on next page)

Corrective action:

1. **The RCS was placed in cold shutdown and drained.**
2. The steam generator was inspected. The tube with the break and adjacent tubes exhibiting wall thinning were plugged. Two springs and part of a hose clamp were found in the steam generator. These were sent to Westinghouse for inspection.

Design purpose of failed system or component:

The steam generators provide the mechanism for removing heat from the RCS.

Unavailability of system per WASH 1400:* —

Unavailability of component per WASH 1400:* **not considered**

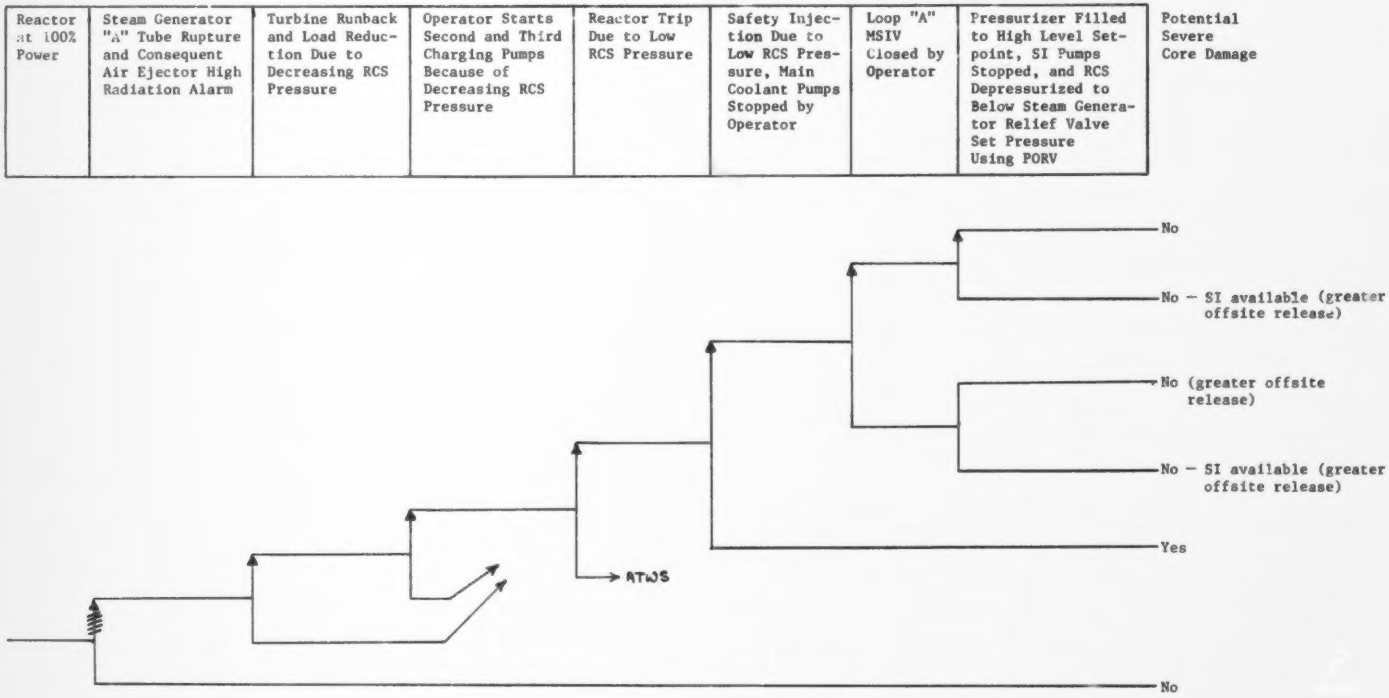
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

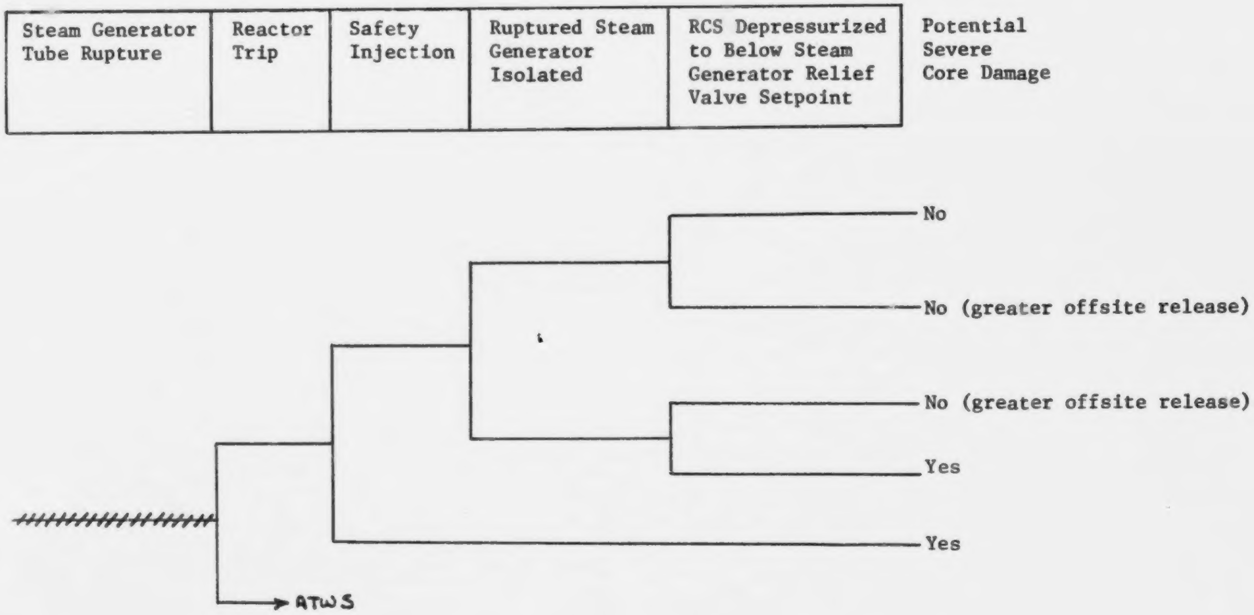
[redacted]

The failure sequence was: (Continued)

7. Steam Generator "A" level increased much more rapidly than usual, indicating a tube break in that generator.
8. Loop "A" Main Steam Isolation Valve closed by operator.
9. No. 11 SI pump was stopped.
10. Operator began depressurizing RCS using PORV.
11. When the pressurizer level reached the high level setpoint the second SI pump was stopped.
12. One hour after the break occurred, RCS pressure was at 910 psig, the same pressure as steam generator "A".


Reactor at 100% Power
Steam Generator "A" Tube Rupture and Consequent Air Ejector High Radiation Alarm
Turbine Runback and Load Reduction Due to Decreasing RCS Pressure
Operator Starts Second and Third Charging Pumps Because of Decreasing RCS Pressure
Reactor Trip Due to Low RCS Pressure
Safety Injection Due to Low RCS Pressure, Main Coolant Pumps Stopped by Operator
Loop "A" MSIV Closed by Operator
Pressurizer Filled to High Level Setpoint, SI Pumps Stopped, and RCS Depressurized to Below Steam Generator Relief Valve Set Pressure Using PORV
Potential Severe Core Damage
No
No – SI available (greater offsite release)
No (greater offsite release)
No – SI available (greater offsite release)
Yes
ATWS
No

NSIC 152563 – Actual Occurrence for Tube Break Occurs in Steam Generator at Prairie Island 1


Steam Generator Tube Rupture
Reactor Trip
Safety Injection
Ruptured Steam Generator Isolated
RCS Depressurized to Below Steam Generator Relief Valve Setpoint
Potential Severe Core Damage
No
No (greater offsite release)
No (greater offsite release)
Yes
Yes
ATWS

NSIC 152563 — Sequence of Interest for Tube Break Occurs at Prairie Island 1

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 152563

DATE OF LER: October 16, 1979

DATE OF EVENT: October 2, 1979

SYSTEM INVOLVED: Reactor Coolant System

COMPONENT INVOLVED: Steam Generator

CAUSE: Tube Break

SEQUENCE OF INTEREST: Steam generator tube rupture

ACTUAL OCCURRENCE: Steam generator tube break

REACTOR NAME: Prairie Island 1

DOCKET NUMBER: 50-282

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 530 MWe

REACTOR AGE: 5.8 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Pioneer Services

OPERATORS: Northern States Power Co.

LOCATION: 28 miles SE of Minneapolis, Minn.

DURATION: N/A

PLANT OPERATING CONDITION: 100% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; (c) made inoperable; [✓ (d)] pressure boundary break

DISCOVERY METHOD: Operational Event

COMMENT: —

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: **152638**

Date: **October 17, 1979**

Title: **Two Auxiliary Feedwater Pumps Concurrently Unavailable at Cook 2**

The failure sequence was:

1. **With the reactor at 100% power, diesel generator "AB" was removed from service for maintenance. This diesel generator provides power to the start and control circuitry for the turbine-driven AFW pump.**
2. **While the "AB" diesel generator was out of service, motor driven AFW pump #1 was also removed from service for maintenance.**

Corrective action:

None - the motor driven pump was returned to service following maintenance.

Design purpose of failed system or component:

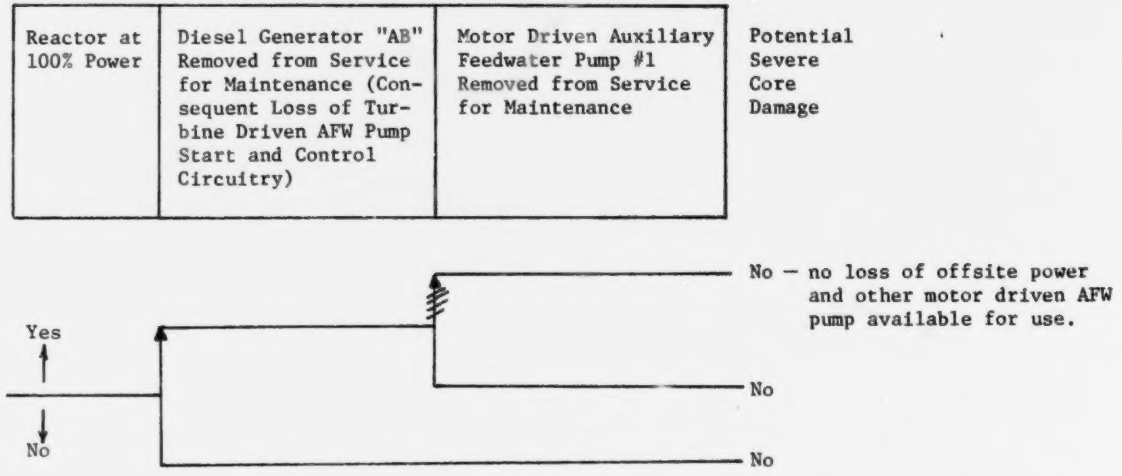
The auxiliary feedwater system provides water to the steam generators for reactor cooling in the event the main feedwater system is unavailable.

Unavailability of system per WASH 1400:*

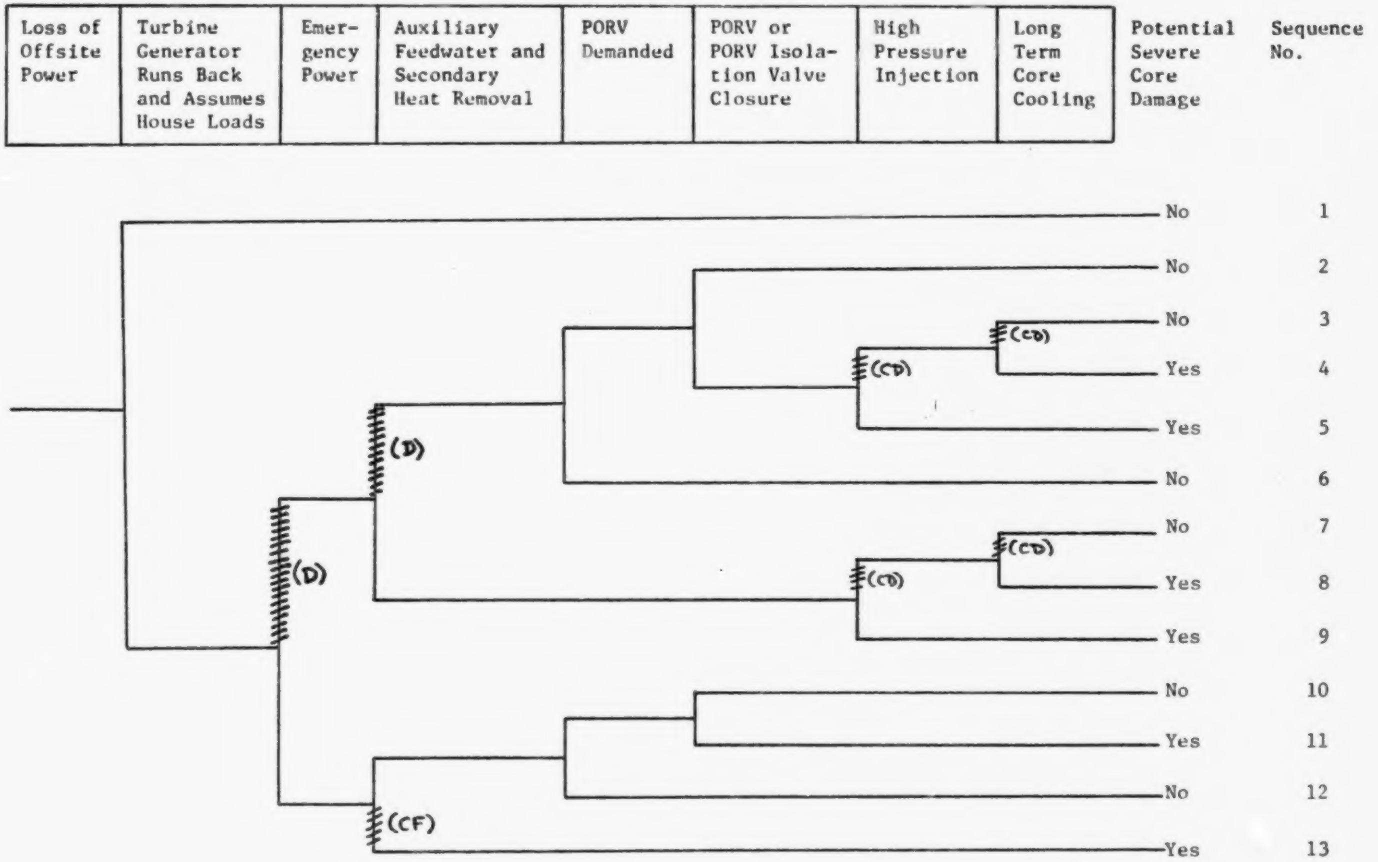
Unavailability of component per WASH 1400:* **motor driven pump: 1×10^{-3}/D**

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor at 100% Power
Diesel Generator "AB" Removed from Service for Maintenance (Consequent Loss of Turbine Driven AFW Pump Start and Control Circuitry)
Motor Driven Auxiliary Feedwater Pump #1 Removed from Service for Maintenance
Potential Severe Core Damage
Yes
No
No — no loss of offsite power and other motor driven AFW pump available for use.
No
No

NSIC 152638 — Actual Occurrence for Two Auxiliary Feedwater Pumps Concurrently Unavailable at Cook 2


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
(D)
(D)
(CD)
(CD)
(CD)
(CD)
(CF)
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
Yes 9
No 10
Yes 11
No 12
Yes 13

NSIC 152638 — Sequence of Interest for Two Auxiliary Feedwater Pumps Concurrently Unavailable at Cook 2

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 152638

DATE OF LER: October 31, 1979

DATE OF EVENT: October 17, 1979

SYSTEM INVOLVED: Auxiliary Feedwater

COMPONENT INVOLVED: Auxiliary feedwater pump

CAUSE: Pump removed from service for maintenance (while a diesel generator was also removed from service for maintenance), human error

SEQUENCE OF INTEREST: Loss of Offsite Power

ACTUAL OCCURRENCE: Removal of an AFW pump from service for maintenance while a diesel generator was also out of service.

REACTOR NAME: Cook 2

DOCKET NUMBER: 50-316

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 1100 MWe

REACTOR AGE: 1.6 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: American Electric Power Service Corp.

OPERATORS: Indiana and Michigan Electric Co.

LOCATION: 11 miles south of Benton Harbor, Michigan

DURATION: 2.75 hours

PLANT OPERATING CONDITION: 100% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ______________________

DISCOVERY METHOD: Operator observation

COMMENT: A 1979 revision to the Cook 1 & 2 FSAR includes a change to the turbine driven AFW pump start and control circuitry to power this circuitry from a separate 250VDC supply. This will eliminate the dependency between the "AB" diesel generator and the turbine-driven AFW pump.

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 152835

Date: July 31, 1979

Title: Engineered Safety Features Actuation System Failed at Calvert Cliffs 1

The failure sequence was:

1. With the reactor at power, trouble-shooting was in progress regarding a fault in the ESFAS Under-voltage (UV) "A" logic circuitry.
2. Both the ESFAS Channel A and B UV Actuation Cabinet Power Supplies were found de-energized due to a trip of the power supply overvoltage protection circuitry.

Corrective action:

1. The power supplies were reset.
2. The over-voltage trip setpoints were raised to prevent future inadvertant trips.
3. Testing was in progress to determine if any common-mode interrelationships existed between the two channels.

Design purpose of failed system or component:

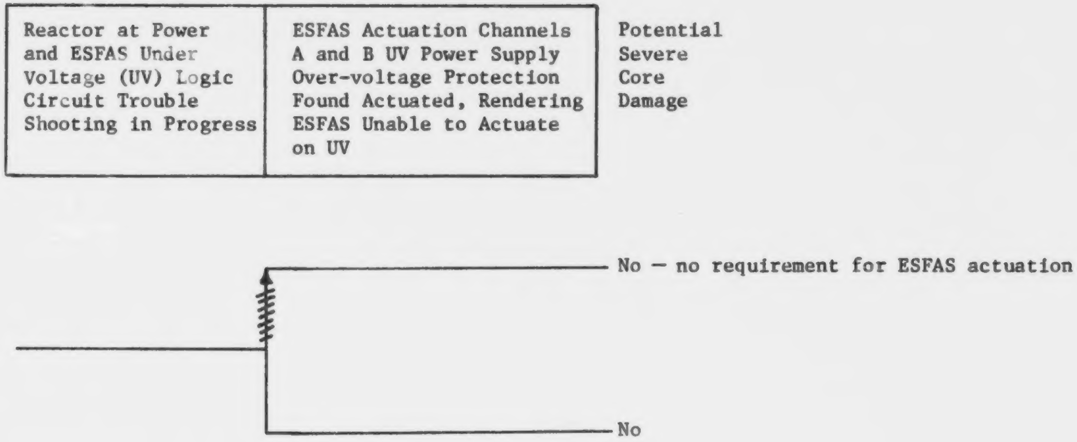
1. The UV Actuation Circuitry sends start commands to the diesel generator and sequencer in the event of a loss of offsite power.

Unavailability of system per WASH 1400:* Under voltage trip, not specifically addressed; Safety Injection Control System total failure: 5.4×10^{-5}/D

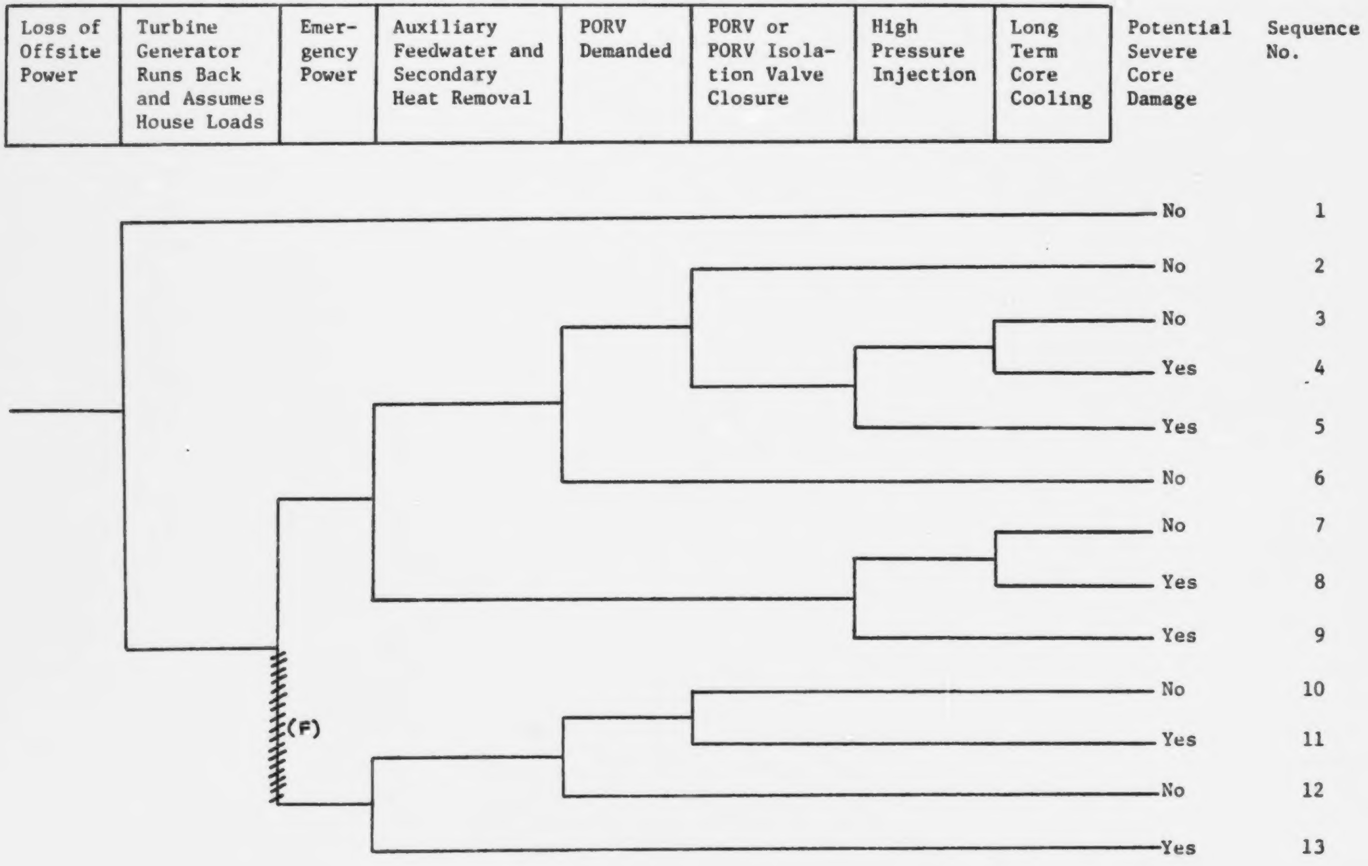
Unavailability of component per WASH 1400:* —

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor at Power and ESFAS Under Voltage (UV) Logic Circuit Trouble Shooting in Progress
ESFAS Actuation Channels A and B UV Power Supply Over-voltage Protection Found Actuated, Rendering ESFAS Unable to Actuate on UV
Potential Severe Core Damage
No — no requirement for ESFAS actuation
No

NSIC 152835 — Actual Occurrence for Engineered Safety Features Actuation System Failed at Calvert Cliffs 1


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
Yes 9
No 10
Yes 11
No 12
Yes 13
(F)

NSIC 152835 — Sequence of Interest for Engineered Safety Features Actuation System Failed at Calvert Cliffs 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 152835

DATE OF LER: August 14, 1979

DATE OF EVENT: July 31, 1979

SYSTEM INVOLVED: Engineered Safety Features Actuation System

COMPONENT INVOLVED: Under-Voltage Actuation Cabinet Power Supplies

CAUSE: De-energized due to trip of over-voltage protection circuitry.

SEQUENCE OF INTEREST: LOOP

ACTUAL OCCURRENCE: Over-voltage protection trip detected during trouble-shooting

REACTOR NAME: Calvert Cliffs 1

DOCKET NUMBER: 50-317

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 845 MWe

REACTOR AGE: 4.8 yr

VENDOR: Combustion Engineering

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Baltimore Gas & Electric Co.

LOCATION: 40 miles south of Annapolis, MD

DURATION: 7.5 hours

PLANT OPERATING CONDITION: At power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start;
[✓ (c)] made inoperable; (d) ______________________

DISCOVERY METHOD: During troubleshooting of UV logic circuits

COMMENT: –

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 152951

Date: October 15, 1979

Title: Offsite Power Lost at Davis-Besse 1

The failure sequence was:

1. A capacitor in an ICS pulser failed due to system voltage spikes above the capacitor working voltage, resulting in the further opening of the turbine control valves.
2. Because of the reduced steam pressure caused by the opened turbine control valves, the ICS attempted to increase reactor power and feedwater flow.
3. The reactor tripped due to high flux at a reduced high flux trip setpoint of ∿69% power.
4. A fault in a generator output breaker caused by a shorted, failed muffler caused a trip of the 345kv "J" bus and a loss of offsite power.
5. Diesel generators No. 1 & 2 started and loaded their safety-related buses.

(See attached page)

Corrective action:

1. The failed ICS capacitor was replaced with one of a higher voltage rating.
2. The damaged generator breaker muffler was replaced and a reinforcing band installed to strengthen all breaker mufflers.
3. The diesel generator output breaker linkage rod cell switch was replaced.

(See attached page)

Design purpose of failed system or component:

Offsite power provides the preferred source of power to safety-related loads when the unit generator is inoperable.

Unavailability of system per WASH 1400:* Offsite power: 2×10^{-5}/hr.

Unavailability of component per WASH 1400:* ICS: not considered in WASH-1400
Circuit Breakers, failure to transfer: 1×10^{-3}/D.

---

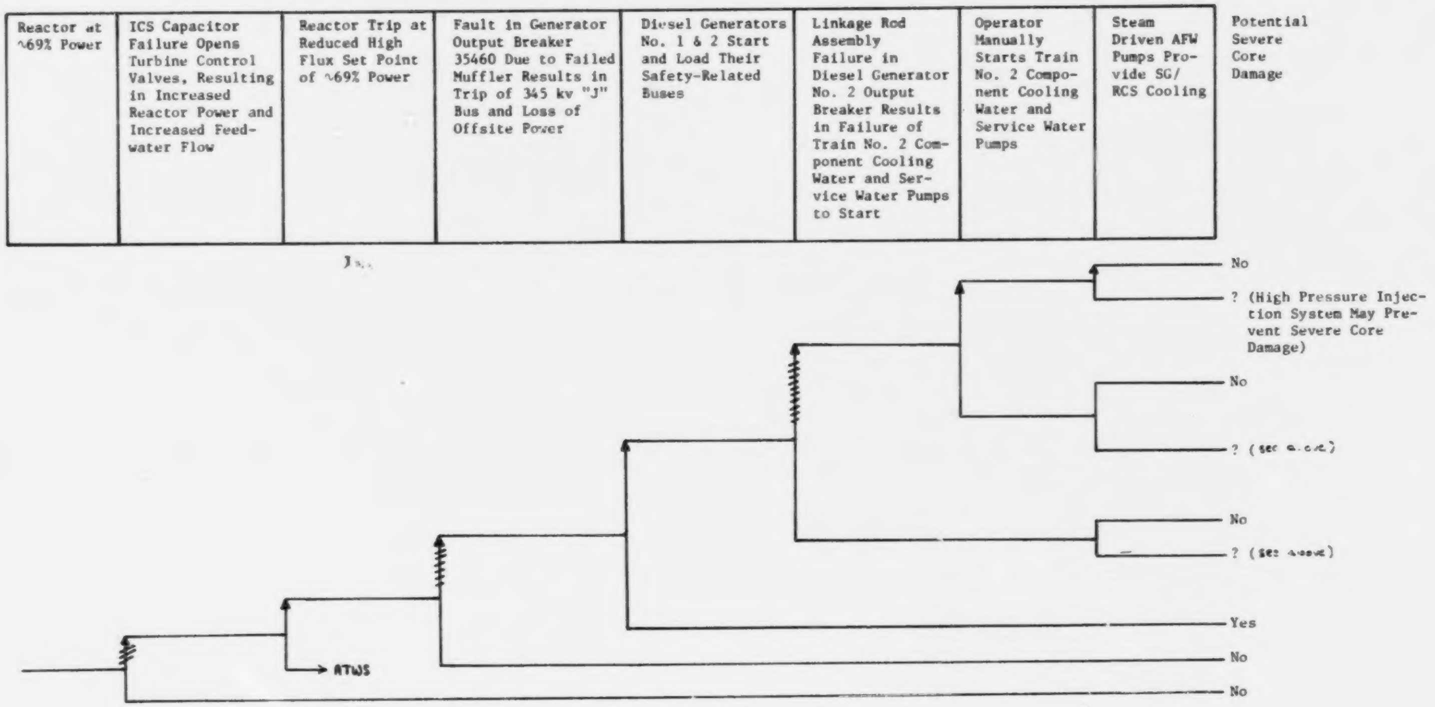
*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

The failure sequence was: (Continued)

6. Due to a linkage-rod failure in the No. 2 diesel generator output breaker, the No. 2 train component cooling water pump and service water pump for diesel cooling did not start. These pumps were manually started.
7. The two steam-driven auxiliary feedwater pumps started and provided steam generator/RCS cooling.
8. During restoration of normal power, sticking motor-cut off switches in the 13.8 kv air circuit breakers for startup transformer No. 2 prevented powering the safety-related buses from offsite power.
9. Operator action successfully closed the startup transformer No. 2 13.8 kv circuit breakers.
10. Defective relays in the Reactor Coolant Pump Component Cooling Water Interlocks prevented restart of Reactor Coolant Pumps 2-2, 1-1, and 2-1.

Corrective action: (Continued)

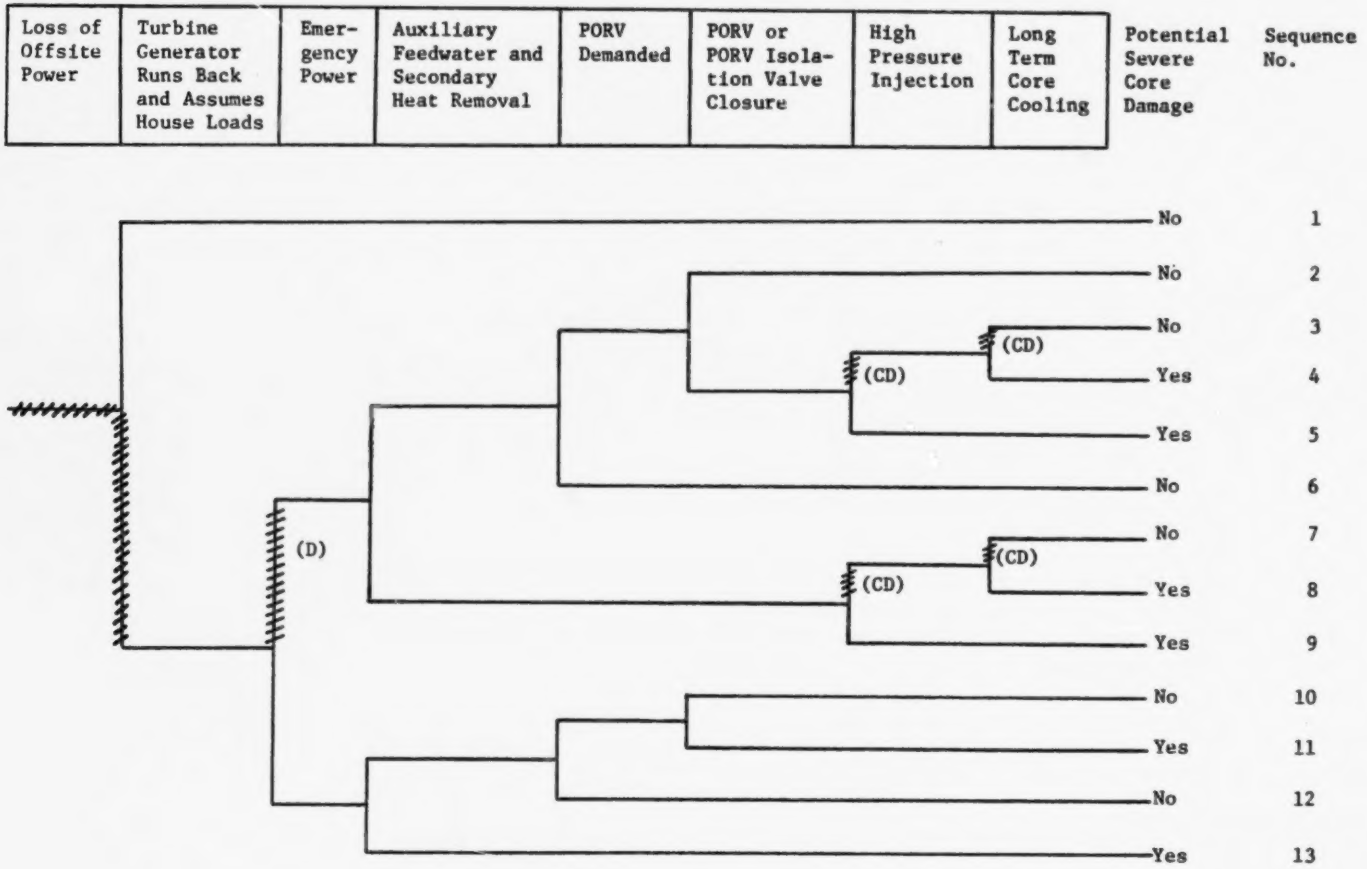
4. The sticking "A" and "B" bus motor cutoff switch operating linkages were replaced.
5. Blown fuses and relays in all reactor coolant pump component cooling water interlocks were replaced.


Reactor at ~69% Power
ICS Capacitor Failure Opens Turbine Control Valves, Resulting in Increased Reactor Power and Increased Feedwater Flow
Reactor Trip at Reduced High Flux Set Point of ~69% Power
Fault in Generator Output Breaker 35460 Due to Failed Muffler Results in Trip of 345 kv "J" Bus and Loss of Offsite Power
Diesel Generators No. 1 & 2 Start and Load Their Safety-Related Buses
Linkage Rod Assembly Failure in Diesel Generator No. 2 Output Breaker Results in Failure of Train No. 2 Component Cooling Water and Service Water Pumps to Start
Operator Manually Starts Train No. 2 Component Cooling Water and Service Water Pumps
Steam Driven AFW Pumps Provide SG/RCS Cooling
Potential Severe Core Damage
No
? (High Pressure Injection System May Prevent Severe Core Damage)
No
? (see above)
No
? (see above)
Yes
ATWS
No
No

NSIC 152951 – Actual Occurrence for Offsite Power Lost at Davis-Besse 1

681

BEST DOCUMENT AVAILABLE


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
(D)
(CD)
(CD)
(CD)
(CD)
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
Yes 9
No 10
Yes 11
No 12
Yes 13

NSIC 152951 — Sequence of Interest for Offsite Power Lost at Davis-Besse 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 152951

DATE OF LER: October 25, 1979

DATE OF EVENT: October 15, 1979

SYSTEM INVOLVED: Integrated Control System (ICS), Offsite Power, Onsite Power

COMPONENT INVOLVED: Capacitor (ICS), Generator Circuit Breaker (Offsite Power), Diesel Output Breaker (Onsite Power)

CAUSE: Capacitor had too low a working voltage, the generator circuit breaker muffler failed and resulted in a ground fault, the Diesel Generator Output breaker linkage rod failed.

SEQUENCE OF INTEREST: Reactor trip with loss of offsite power

ACTUAL OCCURRENCE: Reactor trip with loss of offsite power

REACTOR NAME: Davis-Besse 1

DOCKET NUMBER: 50-346

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 906 MWe

REACTOR AGE: 2.2 yr

VENDOR: B&W

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Toledo Edison Co.

LOCATION: 21 miles east of Toledo, Ohio

DURATION: N/A

PLANT OPERATING CONDITION: at ~69% power

SAFETY FEATURE TYPE OF FAILURE: [✓ (a)] inadequate performance; [✓ (b)] failed to start; [✓ (c)] made inoperable; (d) ________________

DISCOVERY METHOD: During operating

COMMENT: –

683

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 153003

Date: October 24, 1979

Title: LPR Service Water Valves Left Closed Following Testing at Oconee 2

The failure sequence was:

1. Following a revision of the LPR system valve alignment procedure to provide a table of valve positions (to permit double verification of valve position), a LPR system valve alignment was performed.
2. Two days later, with the reactor at 49% power, the LPR cooler service water valves were found to be improperly positioned, due to failure to include the LPR service water valves in the revised check list and failure of the personnel performing the valve check to note that the service water valves were still included in the body of the procedure.

Corrective action:

1. The valves were returned to their correct positions.
2. The LPR procedure was rewritten to include the service water valves in the check list.

Design purpose of failed system or component:

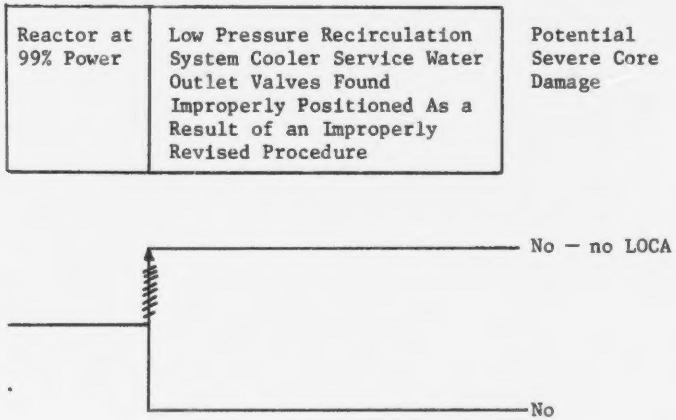
The service water valves provide for the flow of cooling water to the LPR cooler following a loss of coolant accident.

Unavailability of system per WASH 1400:* Low Pressure Recirculation: 1.3×10^{-2}/D

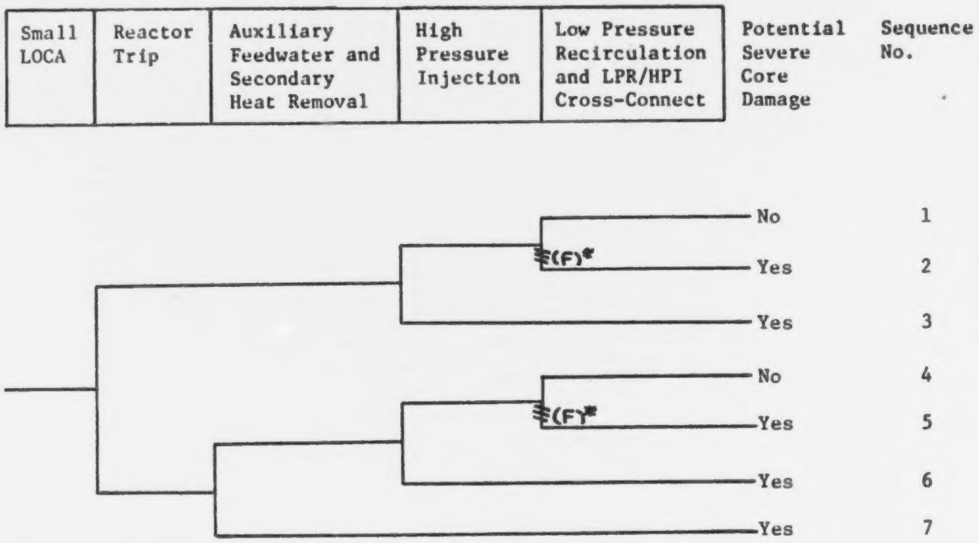
Unavailability of component per WASH 1400:* Valves, motor operated: 1×10^{-3}/D

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor at 99% Power
Low Pressure Recirculation System Cooler Service Water Outlet Valves Found Improperly Positioned As a Result of an Improperly Revised Procedure
Potential Severe Core Damage
No — no LOCA
No

NSIC 153003 — Actual Occurrence for LPI Service Water Valves Left Closed Following a Test at Oconee 2


Small LOCA
Reactor Trip
Auxiliary Feedwater and Secondary Heat Removal
High Pressure Injection
Low Pressure Recirculation and LPR/HPI Cross-Connect
Potential Severe Core Damage
Sequence No.
(F)*
No
1
Yes
2
Yes
3
(F)*
No
4
Yes
5
Yes
6
Yes
7

NSIC 153003 — Sequence of Interest for LPR Service Water Valves Left Closed Following Testing at Oconee 2

[*] Success requires operator detection and opening of closed valves prior to need for LPR.

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 153003

DATE OF LER: November 6, 1979

DATE OF EVENT: October 24, 1979

SYSTEM INVOLVED: Low Pressure Recirculation

COMPONENT INVOLVED: LPR Cooler Service Water Valves

CAUSE: **Valves improperly positioned, human error**

SEQUENCE OF INTEREST: Small LOCA

ACTUAL OCCURRENCE: Failure to Properly Position Valves After Testing

REACTOR NAME: Oconee 2

DOCKET NUMBER: 50-270

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 887 MWe

REACTOR AGE: 6.0 yr

VENDOR: B&W

ARCHITECT-ENGINEERS: DPC/Bechtel

OPERATORS: Duke Power Company

LOCATION: 30 miles west of Greenville, SC

DURATION: 48 hours

PLANT OPERATING CONDITION: 49% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ______________________

DISCOVERY METHOD: Operator observation

COMMENT: –

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 153164

Date: March 28, 1979

Title: Core Damage and Radioactivity Release Occurs at Three Mile Island 2

The failure sequence was:

1. A condensate pump trip while at 99% power resulted in a subsequent trip of both feedwater pumps and a consequent turbine trip and eventual reactor trip.
2. The pressurizer pilot operated relief valve (PORV) opened and stuck open.
3. Auxiliary feedwater flow to the steam generators was blocked by three improperly closed AFW system valves. (AFW was manually initiated 8 minutes after turbine trip.)
4. Decreasing RCS pressure due to the stuck open PORV initiated High Pressure Injection at 1600 psig.
5. Increasing pressurizer level indication caused the operator to infer an increasing pressurizer level. The operator erroneously throttled HPI flow (10-12 minutes after turbine trip).

(see attached sheet)

Corrective action:

1. The stuck open PORV was discovered and its isolation valve closed.
2. The RCS was repressurized and a reactor coolant pump started to provide core cooling.
3. An alternate cooling scheme was installed to permit eventual core cooling without using the decay heat removal system, which would have required pumping highly radioactive reactor coolant outside containment.

Design purpose of failed system or component:

1. The PORV provides for pressure relief for the RCS. It opens before the code safety valves open and prevents their lifting for small pressure excursions.
2. The AFW system provides for core cooling via the steam generators when the main feedwater system is inoperable.
3. The HPI system provides high pressure borated water to the RCS in the event of a small break in the reactor coolant system.

Unavailability of system per WASH 1400:* AFW: 2.5×10^{-4}/D (start + 8 hours)
HPI: 1.2×10^{-2}/D

Unavailability of component per WASH 1400:* PORV, failure to reseat: 10^{-2}/D
general human error: 5×10^{-3}/D

---

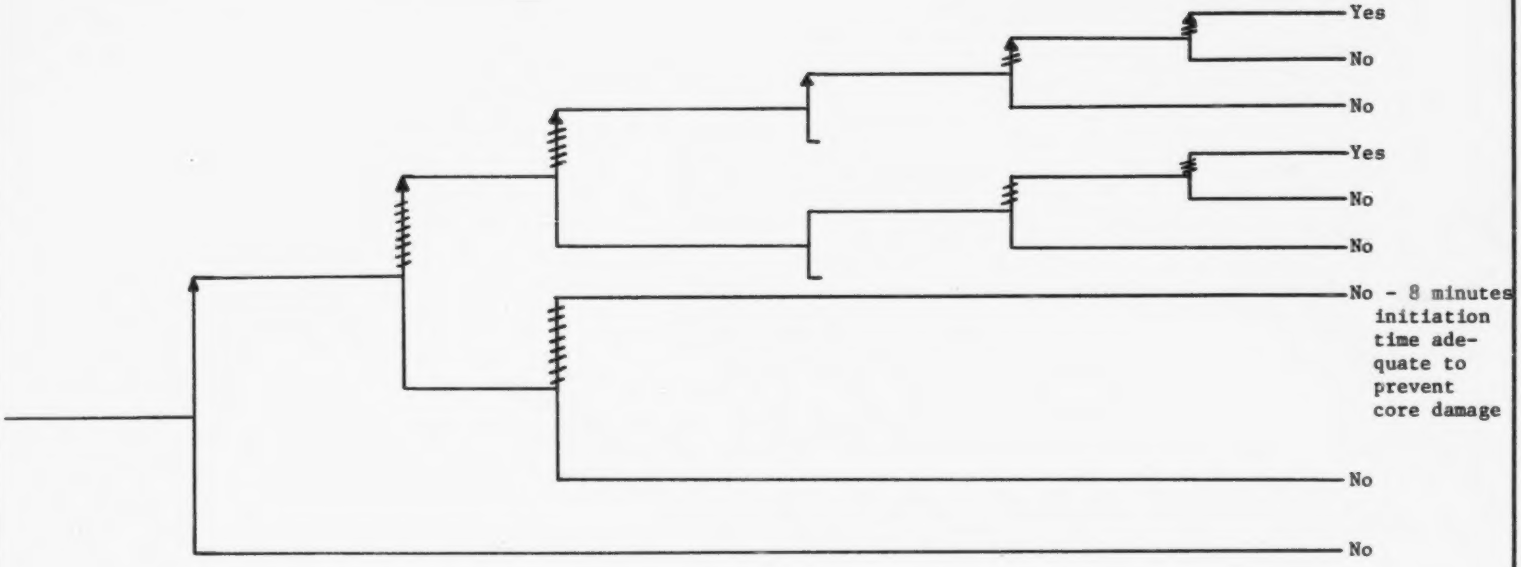
*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

[redacted]

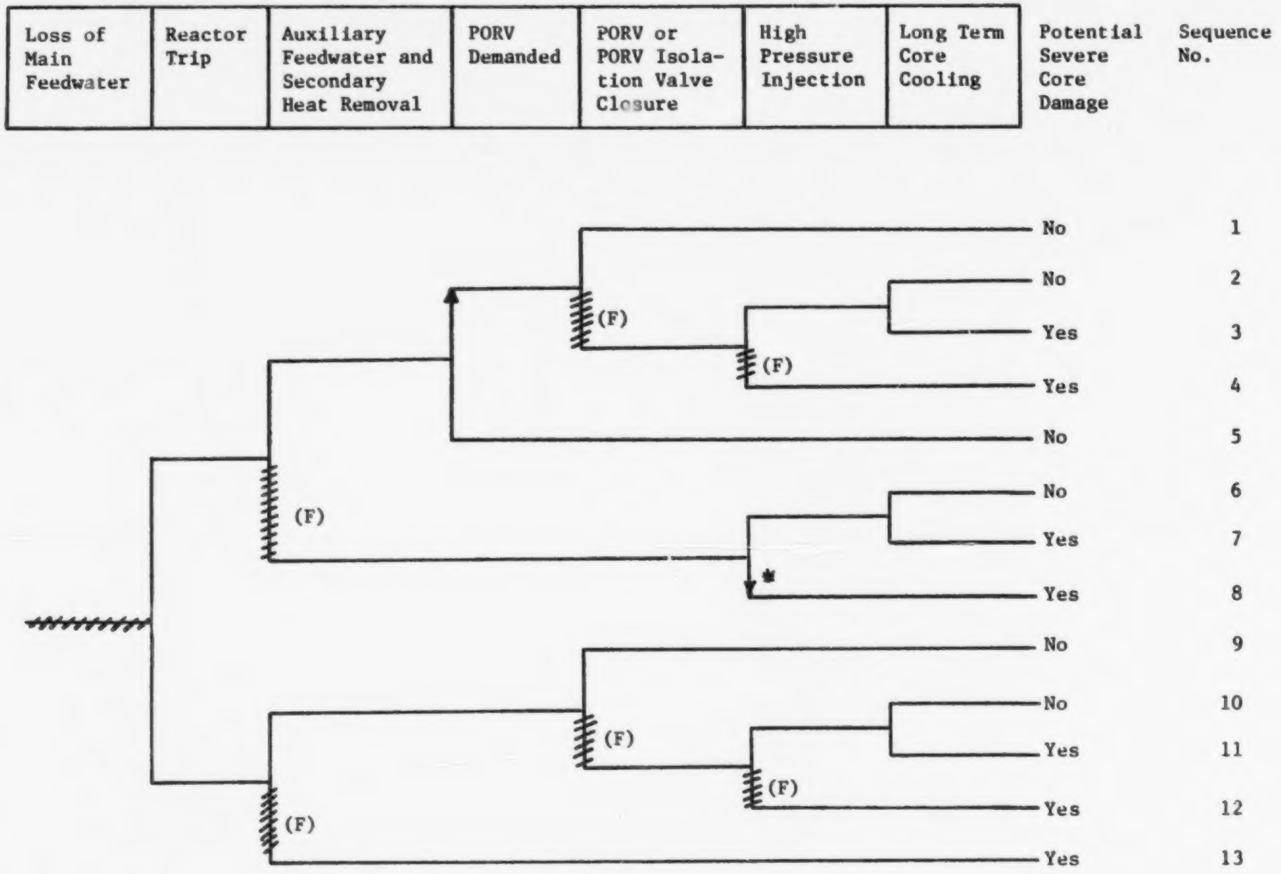
The failure sequence was: (continued)

6. Between one and two hours after turbine trip, the operator stopped all Reactor Coolant Pumps because of pump vibration. Due to the loss of forced core cooling and the inability to cool the core by natural circulation because of voids which existed in the RCS as a result of inadequate HPI flow, core damage occurred.

Reactor at 99% power and condensate pump trip	Main feedwater pump trip, loss of feedwater and consequent turbine/reactor trip	PORV lifts and sticks open	Auxiliary feedwater flow temporarily unavailable due to closed AFW valves (8 minutes)	Decreasing RCS pressure results in HPI initiation	Increasing pressurizer level indication causes operator to infer an increasing pressurizer level and throttle back HPI	Operator stops reactor coolant pumps because of pump vibration	Potential Severe Core Damage


Yes
No
No
Yes
No
No
No - 8 minutes initiation time adequate to prevent core damage
No
No

NSIC 153164 — Actual Occurrence for Core Damage and Radioactivity Release Occurs at Three Mile Island 2


Loss of Main Feedwater
Reactor Trip
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
(F)
(F)
(F)
(F)
(F)
(F)
(F)
*
No 1
No 2
Yes 3
Yes 4
No 5
No 6
Yes 7
Yes 8
No 9
No 10
Yes 11
Yes 12
Yes 13

NSIC 153164 — Sequence of Interest for Core Damage and Radioactivity Release Occurs at Three Mile Island 2
*Not included in mitigation procedures.

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 153164

DATE OF LER:

DATE OF EVENT: March 28, 1979

SYSTEM INVOLVED: reactor coolant system, auxiliary feedwater, high pressure injection

COMPONENT INVOLVED: pilot operated relief valve

CAUSE: stuck open PORV, isolated AFW system, operator errors related to high pressure injection and RC pumps, human error

SEQUENCE OF INTEREST: loss of feedwater

ACTUAL OCCURRENCE: loss of feedwater and subsequent failed open PORV, degraded AFW system, failed high pressure injection, and RC pump trip result in core damage.

REACTOR NAME: Three Mile Island 2

DOCKET NUMBER: 50-320

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 906 MWe

REACTOR AGE: 1.0 yr

VENDOR: B&W

ARCHITECT-ENGINEERS: Burns & Roe

OPERATORS: Metropolitan Edison

LOCATION: 10 mi SE of Harrisburg, Pa.

DURATION: N/A

PLANT OPERATING CONDITION: 99% power

SAFETY FEATURE TYPE OF FAILURE: [✓ (a)] inadequate performance; [✓ (b)] failed to start; [✓ (c)] made inoperable; (d) ____________________

DISCOVERY METHOD: operational event

COMMENT: A formal LER on this event has not been received at NSIC.

BEST DOCUMENT AVAILABLE

692

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 153167

Date: September 26, 1979

Title: Gas Turbine Power Supplies Fail at Oconee

The failure sequence was:

1. With Units 1 and 2 at approximately 100% power, both Keowee Hydro units were removed from service for repair and a Lee Steam Station gas turbine started to provide power to the standby buses.
2. The Lee Station gas turbine repeatedly tripped during operation, resulting in unavailability of power to the standby buses.

Corrective action:

Investigation revealed no reason for the gas turbine's tripping. Procedures were revised to require immediate shutdown of the Oconee Units in the event the Keowee Hydro Units are out of service and the Lee Station Turbine trips.

Design purpose of failed system or component:

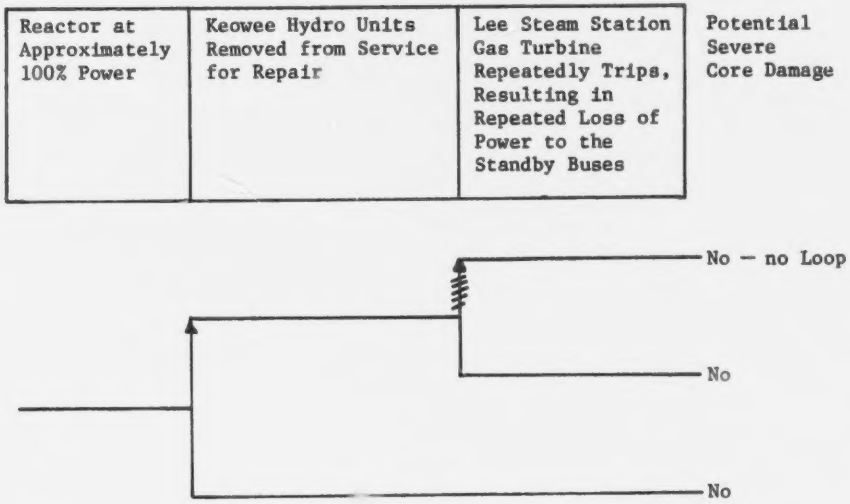
The Lee Steam Station Gas Turbines are used to provide power to the Oconee standby buses in the event the normal source of emergency power, the Keowee Hydro units, are removed from service for maintenance.

Unavailability of system per WASH 1400:* on-site electrical power: 1×10^{-2}/D

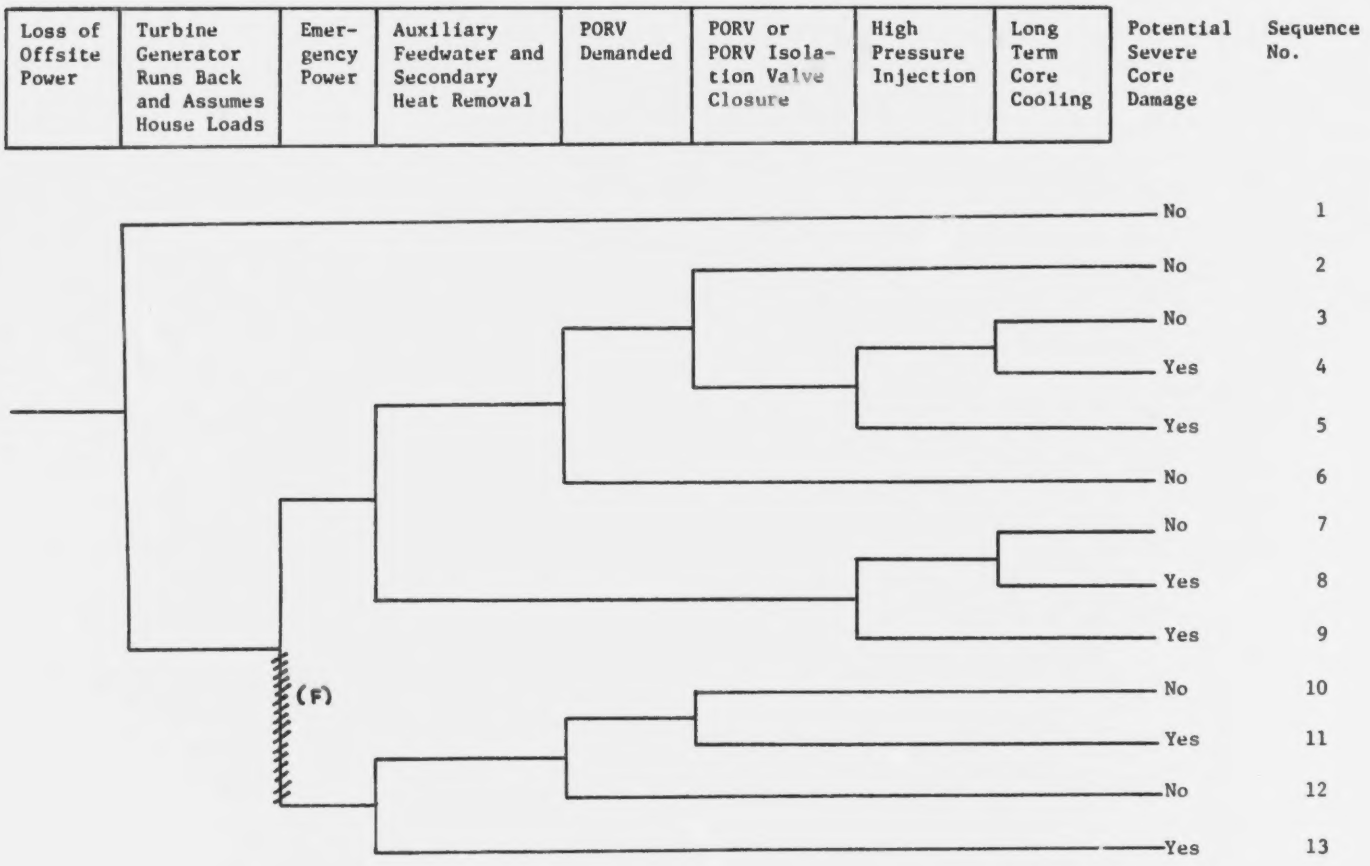
Unavailability of component per WASH 1400:* not considered

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}


Reactor at Approximately 100% Power
Keowee Hydro Units Removed from Service for Repair
Lee Steam Station Gas Turbine Repeatedly Trips, Resulting in Repeated Loss of Power to the Standby Buses
Potential Severe Core Damage
No — no Loop
No
No

NSIC 153167 — Actual Occurrence for Gas Turbine Power Supplies Fail at Oconee


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
Yes 9
No 10
Yes 11
No 12
Yes 13
(F)

NSIC 153167 — Sequence of Interest for Gas Turbine Power Supplies Fail at Oconee

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 153167

DATE OF LER: October 1, 1979

DATE OF EVENT: September 26, 1979

SYSTEM INVOLVED: emergency power

COMPONENT INVOLVED: gas turbine generator

CAUSE: repeated generator trips

SEQUENCE OF INTEREST: loss of offsite power

ACTUAL OCCURRENCE: loss of standby power source during power operation

REACTOR NAME: Oconee 1 and 2 (Oconee 3 in cold shutdown)

DOCKET NUMBER: 50-269 and 50-270

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 887 MWe

REACTOR AGE: 6.4 yr (Oconee 1)

VENDOR: B&W

ARCHITECT-ENGINEERS: DPC/Bechtel

OPERATORS: Duke Power Co.

LOCATION: 30 miles west of Greenville, SC

DURATION: 2 hours

PLANT OPERATING CONDITION: both plants at approximately 100% power

SAFETY FEATURE TYPE OF FAILURE: [✓ (a)] inadequate performance; (b) failed to start; (c) made inoperable; (d) ____________________

DISCOVERY METHOD: operational event

COMMENT: —

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: **153333**

Date: **November 15, 1979**

Title: **Flowpath from AFW Pump to Two Steam Generators Made Inoperable at Cook 1**

The failure sequence was:

With the reactor at 100% power and the turbine driven AFW pump out of service for repair, the motor control center which provides power for valves which direct flow from one motor-driven AFW pump to two of the four steam generators was removed from service for inspection.

Corrective action:

The motor control center was returned to service.

Design purpose of failed system or component:

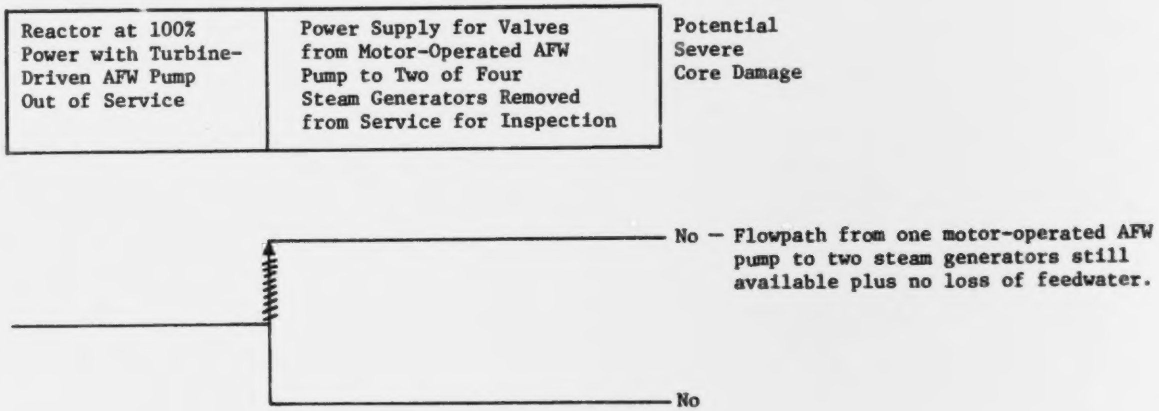
The motor control center provides power to the two motor operated valves associated with the AFW flow path from the motor-driven AFW pumps to two of the four steam generators.

Unavailability of system per WASH 1400:* **AFW (loss of net + 8 hrs): 2.5×10^{-4}**

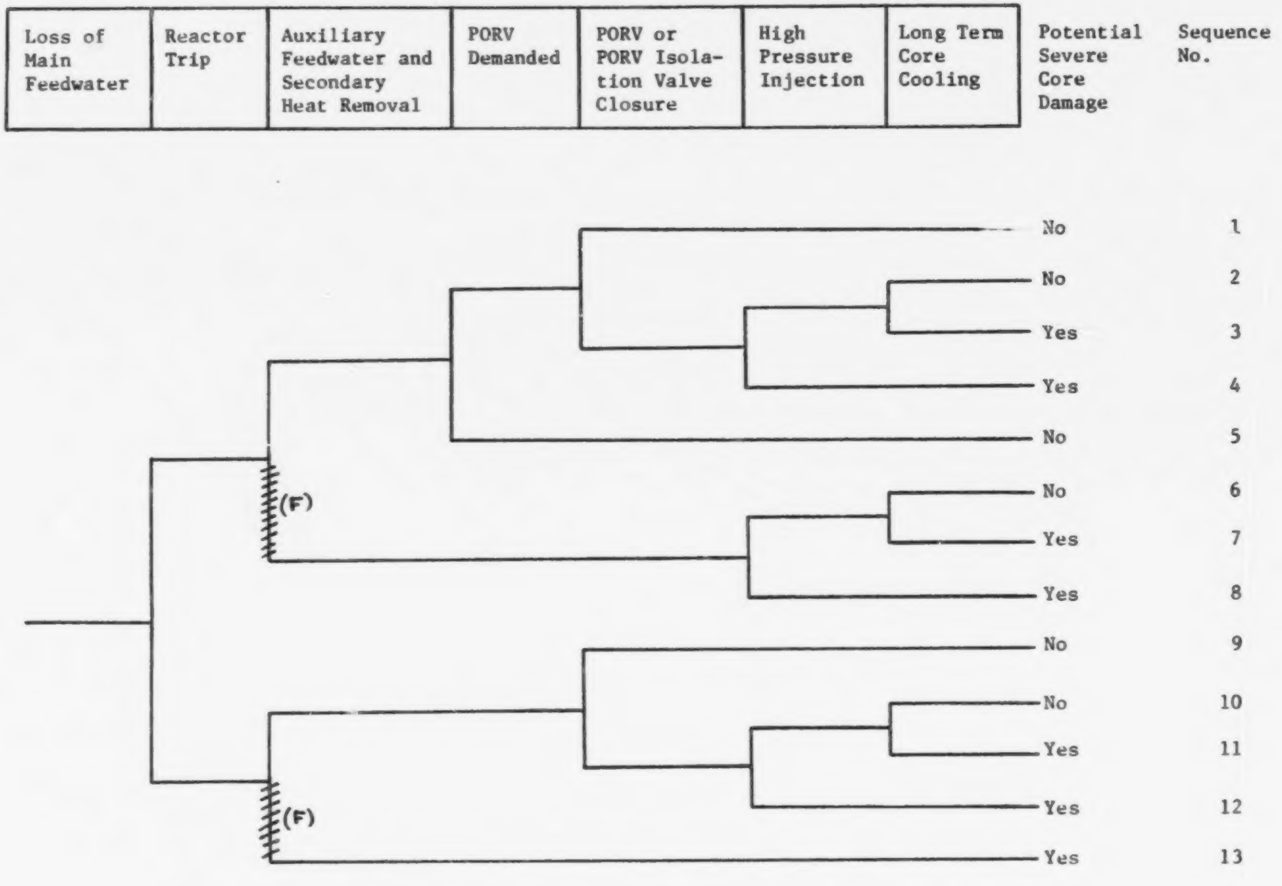
Unavailability of component per WASH 1400:* —

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor at 100% Power with Turbine-Driven AFW Pump Out of Service
Power Supply for Valves from Motor-Operated AFW Pump to Two of Four Steam Generators Removed from Service for Inspection
Potential Severe Core Damage
No — Flowpath from one motor-operated AFW pump to two steam generators still available plus no loss of feedwater.
No

NSIC 153333 — Actual Occurrence for Flowpath from AFW Pump to Two Steam Generators Made Inoperable at Cook 1


Loss of Main Feedwater
Reactor Trip
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
(F)
(F)
No 1
No 2
Yes 3
Yes 4
No 5
No 6
Yes 7
Yes 8
No 9
No 10
Yes 11
Yes 12
Yes 13

NSIC 153333 — Sequence of Interest for Flowpath from AFW Pump to Two Steam Generators Made Inoperable at Cook 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 153333

DATE OF LER: November 29, 1979

DATE OF EVENT: November 15, 1979

SYSTEM INVOLVED: Auxiliary Feedwater

COMPONENT INVOLVED: Motor Operated Valves

CAUSE: Motor Control Center removed from service for inspection while the turbine-driven AFW pump was out of service for repair, human error

SEQUENCE OF INTEREST: loss of of feedwater

ACTUAL OCCURRENCE: unavailability of two AFW system valves plus the turbine driven AFW pump

REACTOR NAME: Cook 1

DOCKET NUMBER: 50-315

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 1054 MWe

REACTOR AGE: 4.8 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: American Electric Power Corp.

OPERATORS: Indiana and Michigan Electric Co.

LOCATION: 11 miles south of Benton Harbor, Michigan

DURATION: 4 hours

PLANT OPERATING CONDITION: 100% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________

DISCOVERY METHOD: operator observation

COMMENT: The Cook 1 FSAR states the LOFW analysis for Cook 1 assumes flow to all four steam generators. The failure reported herein resulted in flow to only two steam generators.

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 153338

Date: November 1, 1979

Title: Maintenance Performed on Wrong Diesel Generator at Cook 2

The failure sequence was:

1. The reactor was in a refueling shutdown and one diesel generator was out of service due to installation of a design change.
2. Contractor personnel mistakenly began to terminate the operable diesel generator's starting bus feed cable.
3. This resulted in a short circuit and blown fuse in the diesel generator inverter.

Corrective action:

1. The inverter was switched to an alternate source until the fuse was replaced.

Design purpose of failed system or component:

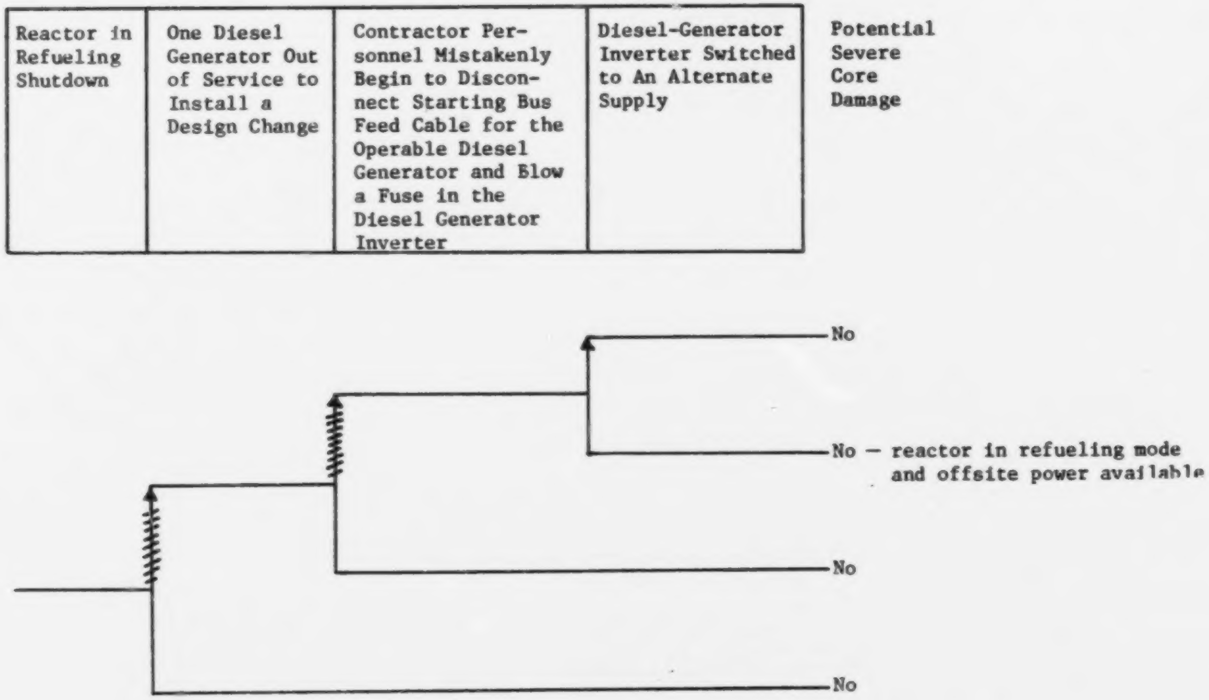
The diesel-generator provides a source of on-site emergency power when the main generator and both offsite power sources are unavailable.

Unavailability of system per WASH 1400:* electric power: 1×10^{-2}/D

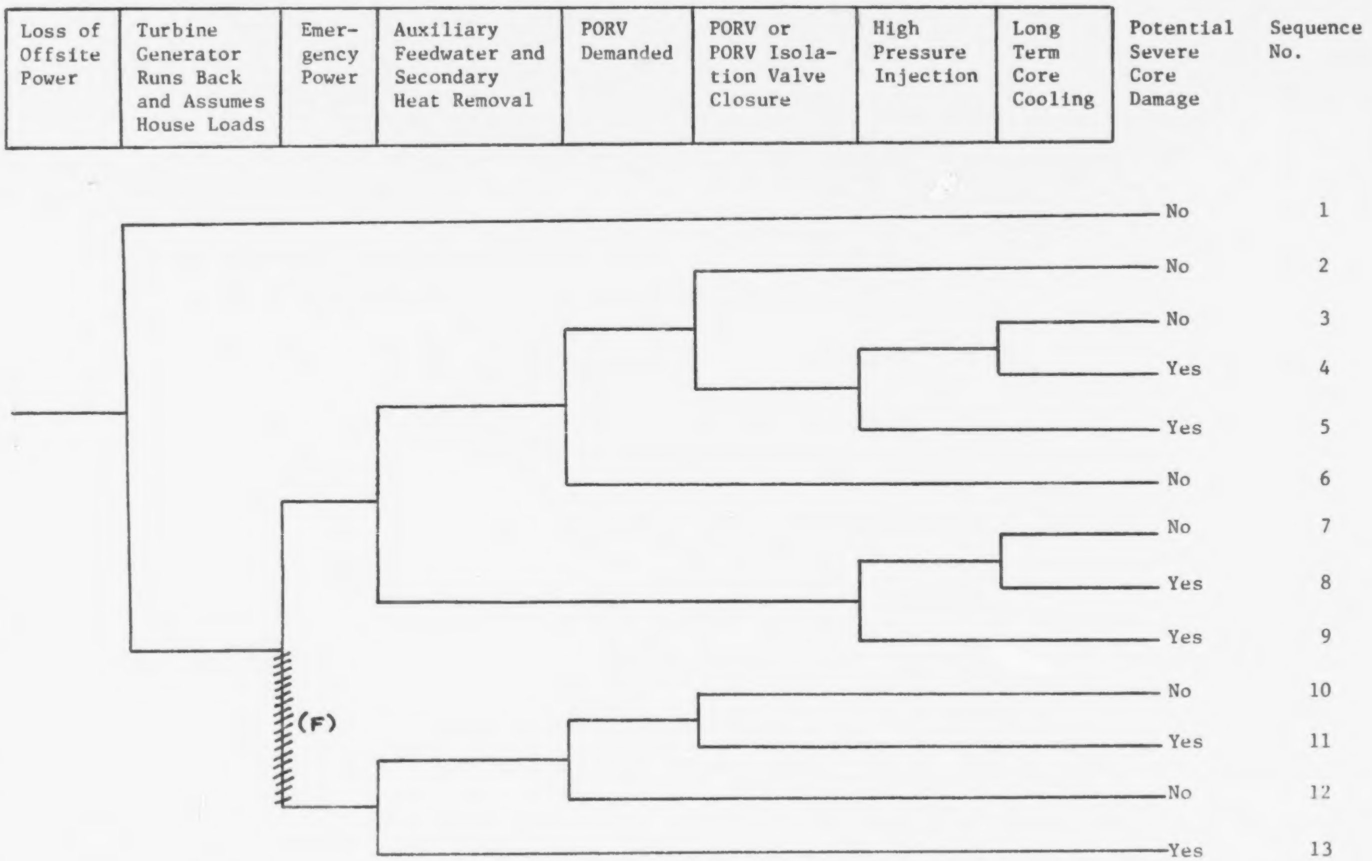
Unavailability of component per WASH 1400:* diesel generator: 3×10^{-2}/D

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour hR^{-1}.


Reactor in Refueling Shutdown
One Diesel Generator Out of Service to Install a Design Change
Contractor Personnel Mistakenly Begin to Disconnect Starting Bus Feed Cable for the Operable Diesel Generator and Blow a Fuse in the Diesel Generator Inverter
Diesel-Generator Inverter Switched to An Alternate Supply
Potential Severe Core Damage
No
No — reactor in refueling mode and offsite power available
No
No

NSIC 153338 — Actual Occurrence for Maintenance Performed on Wrong Diesel Generator at Cook 2


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
Yes 9
No 10
Yes 11
No 12
Yes 13
(F)

NSIC 153338 — Sequence of Interest for Maintenance Performed on Wrong Diesel Generator at Cook 2

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 153338

DATE OF LER: December 3, 1979

DATE OF EVENT: November 1, 1979

SYSTEM INVOLVED: Electric power

COMPONENT INVOLVED: Diesel generator

CAUSE: Maintenance was performed on the operable diesel-generator in lieu of the out-of-service diesel generator, human error

SEQUENCE OF INTEREST: Loss of offsite power with reactor at power

ACTUAL OCCURRENCE: Incorrect diesel generator maintenance during refueling.

REACTOR NAME: Cook 2

DOCKET NUMBER: 50-316

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 1100 MWe

REACTOR AGE: 1.6 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: American Electric Power Service Corp.

OPERATORS: Indiana and Michigan Electric Co.

LOCATION: 11 miles south of Benton Harbor, Michigan

DURATION: .75 hours

PLANT OPERATING CONDITION: Refueling

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c) made inoperable]; (d) ____________

DISCOVERY METHOD: Alarm during refueling

COMMENT: –

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 153686

Date: November 27, 1979

Title: Redundant High Head Safety Injection Pumps Made Inoperable at Beaver Valley 1

The failure sequence was:

1. With the reactor at 35% power, diesel generator No. 1 was removed from service for preventive maintenance.
2. While the diesel generator was unavailable, motor operated valve CH-115D was removed from service for maintenance. This, combined with the unavailability of the diesel generator, would result in a failure to deliver water to the high head safety injection pumps from the RWST in the event of a loss of offsite power.

Corrective action:

The closed valve was reopened.

Design purpose of failed system or component:

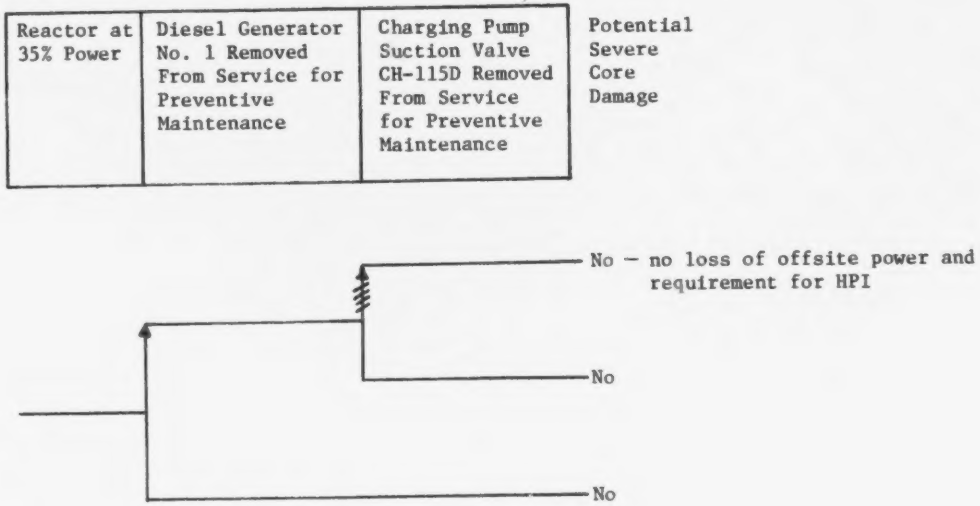
Motor operated valve CH-115D is one of two normally closed parallel valves which provide water from the RWST to the high head safety injection/charging pumps.

Unavailability of system per WASH 1400:* High Pressure Injection: 1.2×10^{-2} (Two of Three Pumps Required)

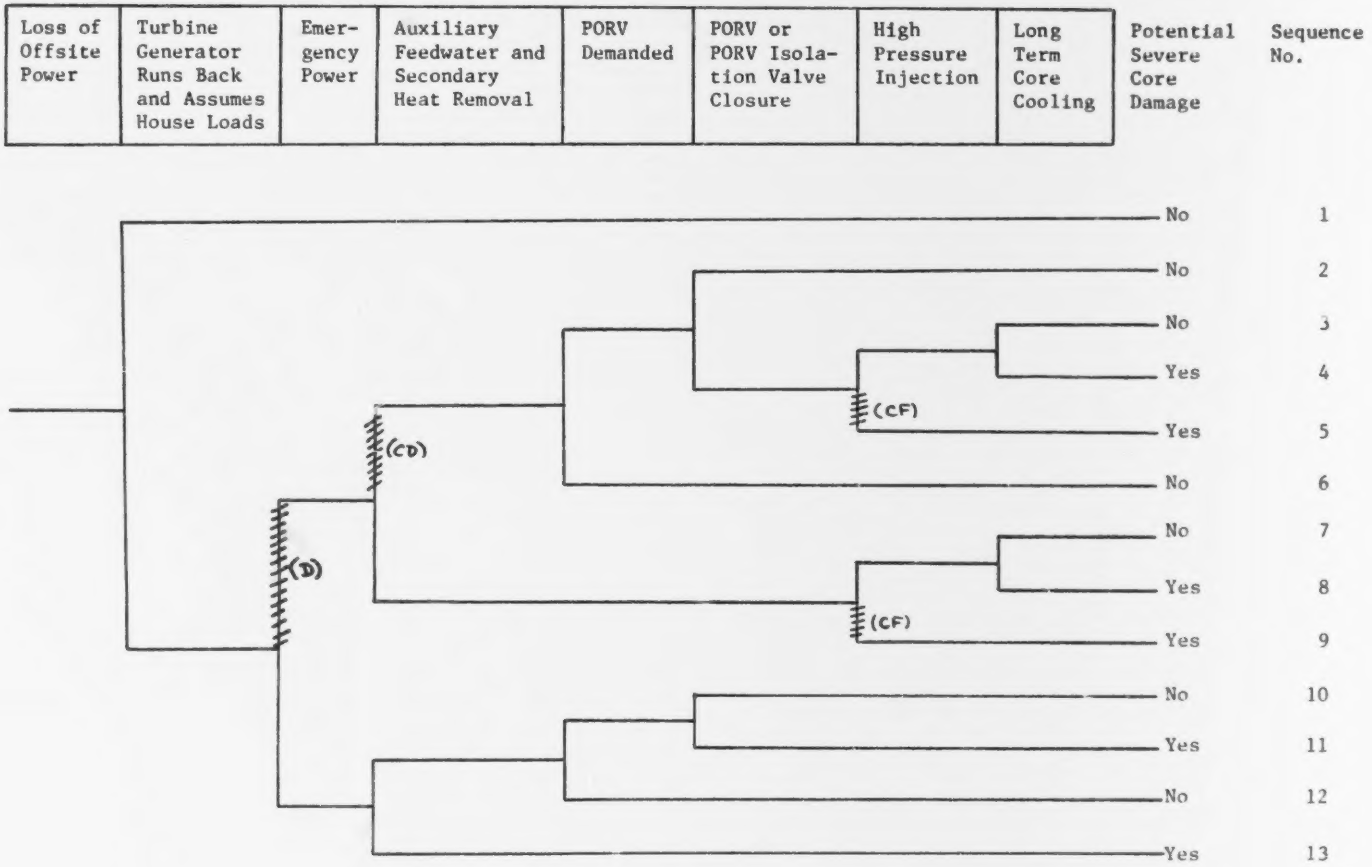
Unavailability of component per WASH 1400:* Diesel generator: 3×10^{-2}/D
Motor operated valve: 1×10^{-3}/D

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor at 35% Power
Diesel Generator No. 1 Removed From Service for Preventive Maintenance
Charging Pump Suction Valve CH-115D Removed From Service for Preventive Maintenance
Potential Severe Core Damage
No – no loss of offsite power and requirement for HPI
No
No

NSIC 153686 – Actual Occurrence for Redundant High Head Safety Injection Pumps Made Inoperable at Beaver Valley 1


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
Yes 9
No 10
Yes 11
No 12
Yes 13
(D)
(CD)
(CF)
(CF)

NSIC 153686 — Sequence of Interest for Redundant High Head Safety Injection Pumps Made Inoperable at Beaver Valley 1

707

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 153686

DATE OF LER: December 11, 1979

DATE OF EVENT: November 27, 1979

SYSTEM INVOLVED: Emergency power, high head safety injection

COMPONENT INVOLVED: Diesel generator, motor operated valve, human error

CAUSE: While a diesel generator was removed from service for preventive maintenance, a charging pump suction valve was inadvertently closed for maintenance.

SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE: Diesel generator and charging pump suction valve concurrently removed from service for maintenance.

REACTOR NAME: Beaver Valley 1

DOCKET NUMBER: 50-334

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 852 MWe

REACTOR AGE: 3.6 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Stone & Webster

OPERATORS: Duquesne Light Co.

LOCATION: 5 miles east of East Liverpool, Ohio

DURATION: 1.75 hours

PLANT OPERATING CONDITION: at 35% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________________

DISCOVERY METHOD: Operator observation

COMMENT: –

[redacted]

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 153810

Date: November 20, 1979

Title: RCIC Turbine Trips with HPCI Unavailable at Brunswick 2

The failure sequence was:

1. Reactor trip occurred. (reason unspecified)
2. HPCI was unavailable due to maintenance.
3. RCIC initiated on low-low reactor water level (-38 inches).
4. The RCIC turbine tripped on overspeed.
5. RCIC was reset and manually set into operation. The reactor water level had reached -40 inches.

Corrective action:

An investigation revealed a crack in the amphenal connector to the tachometer pickup causing a loose connection. This connector was replaced and the turbine tested with satisfactory results.

Design purpose of failed system or component:

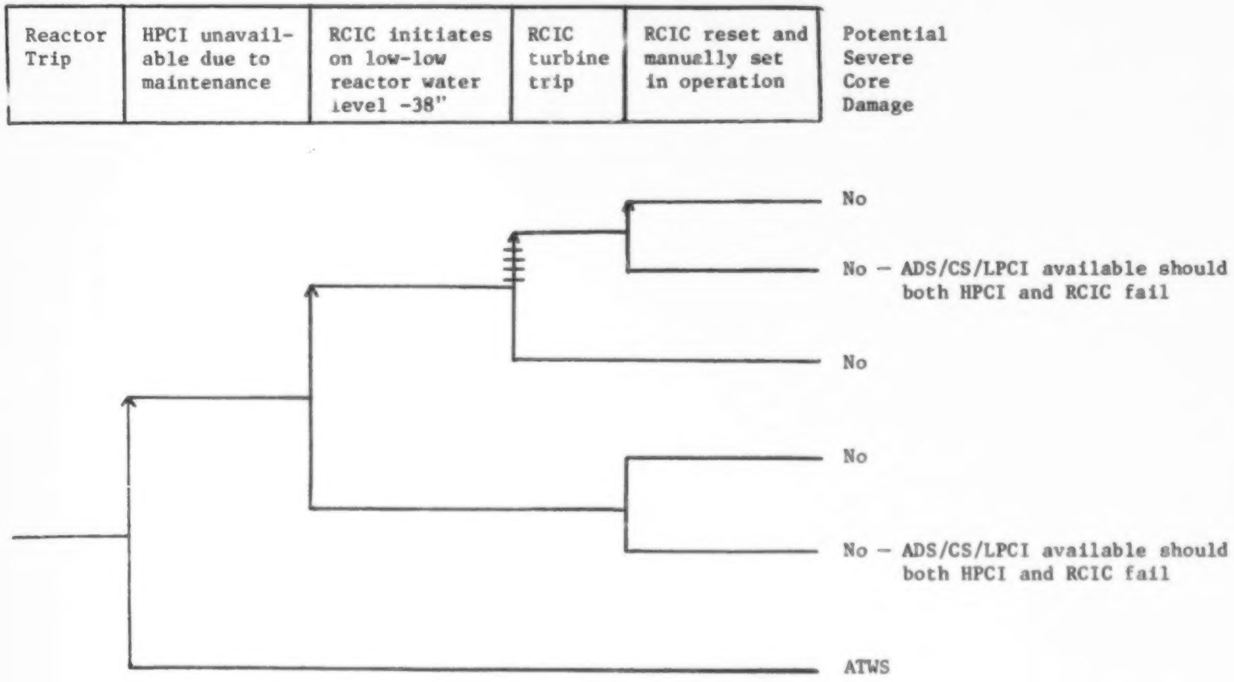
1. The RCIC provides reactor core cooling and water level control during a loss of feedwater flow.
2. The tachometer monitors the turbine speed for control purposes.

Unavailability of system per WASH 1400:* RCIC: $8 \times 10^{-2}/D$

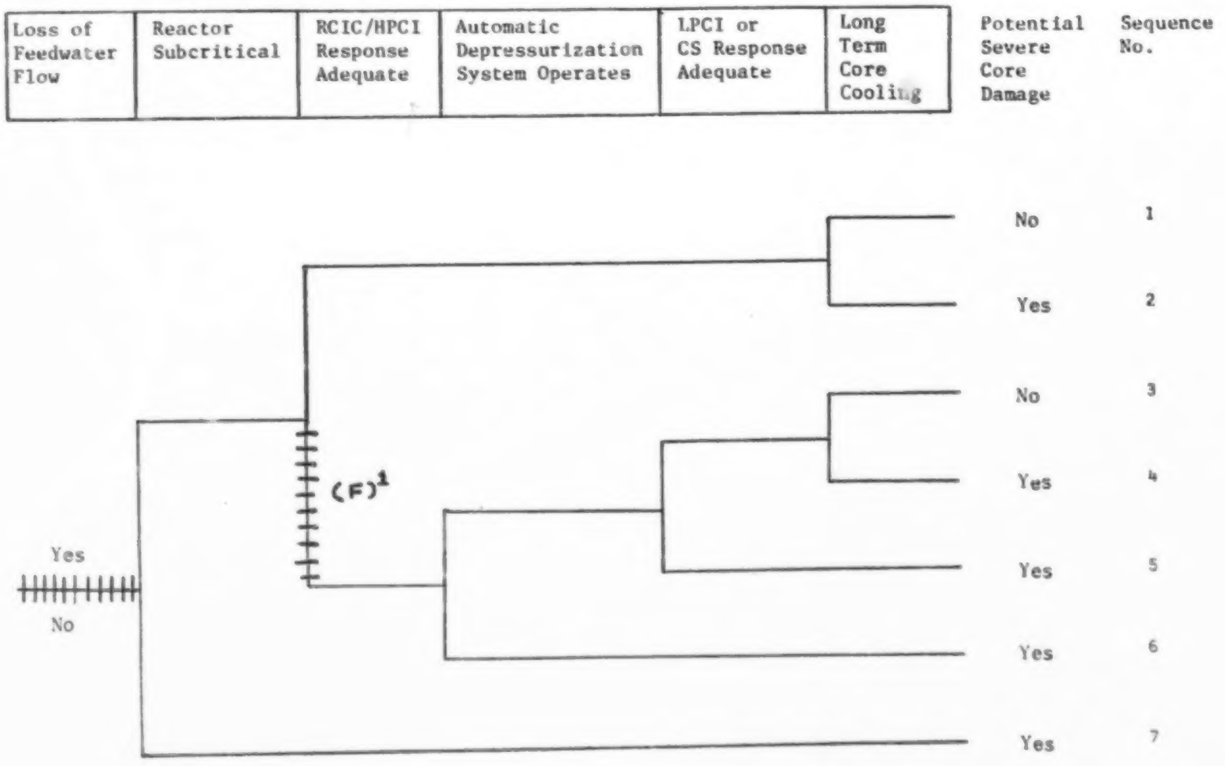
Unavailability of component per WASH 1400:* –

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor Trip
HPCI unavailable due to maintenance
RCIC initiates on low-low reactor water level -38"
RCIC turbine trip
RCIC reset and manually set in operation
Potential Severe Core Damage
No
No — ADS/CS/LPCI available should both HPCI and RCIC fail
No
No
No — ADS/CS/LPCI available should both HPCI and RCIC fail
ATWS

NSIC 153810 — Actual Occurrence of RCIC Turbine Trip with HPCI Unavailable at Brunswick 1


Loss of Feedwater Flow
Reactor Subcritical
RCIC/HPCI Response Adequate
Automatic Depressurization System Operates
LPCI or CS Response Adequate
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
Yes
No
(F)[1]
No 1
Yes 2
No 3
Yes 4
Yes 5
Yes 6
Yes 7

NSIC 153810 — Sequence of Interest for RCIC Turbine Trip with HPCI Unavailable at Brunswick 1

[1]Success requires the operator to reset and manually start RCIC.

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 153810

DATE OF LER: December 17, 1979

DATE OF EVENT: November 20, 1979

SYSTEM INVOLVED: RCIC

COMPONENT INVOLVED: Tachometer

CAUSE: Mechanical failure

SEQUENCE OF INTEREST: Loss of feedwater flow

ACTUAL OCCURRENCE: RCIC turbine trip with HPCI unavailable

REACTOR NAME: Brunswick 1

DOCKET NUMBER: 50-325

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 821 MWe

REACTOR AGE: 3.11 yr

VENDOR: GE

ARCHITECT-ENGINEERS: United Engineers and Constructors

OPERATORS: Carolina Power and Light Co.

LOCATION: Three miles N of Southport, NC

DURATION: N/A

PLANT OPERATING CONDITION: Scrammed

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________________

DISCOVERY METHOD: operator surveillance

COMMENT: —

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 154286

Date: November 28, 1979

Title: Safety Injection Pump Fails at Salem 1

The failure sequence was:

1. With the reactor in cold shutdown, High Pressure Safety Injection Pump No. 12 was started to refill an accumulator.
2. The pump breaker tripped when the pump started.
3. Investigation revealed the impeller locknuts on the pump had loosened, resulting in worn spacer sleeves and a bent impeller.
4. Further investigation revealed the impeller lock nuts on the other High Pressure Injection Pump were also loose, as were the lock nuts on one pump in the other unit.

Corrective action:

The damaged rotating element was temporarily replaced with one from a pump in the other unit. The lock nuts were secured using Locktite.

Design purpose of failed system or component:

The high pressure injection pumps provide the source of high pressure injection used for loss of coolant accident mitigation.

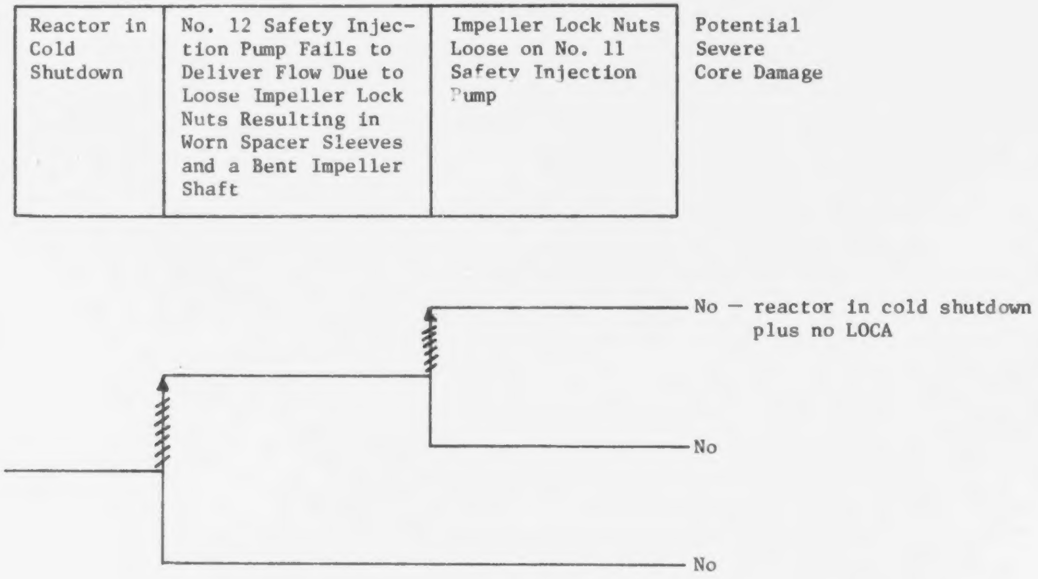
Unavailability of system per WASH 1400:* High Pressure Injection (2 of 3 pumps): 1.2×10^{-2}

Unavailability of component per WASH 1400:* Pump, failure to start: 1×10^{-3}/D

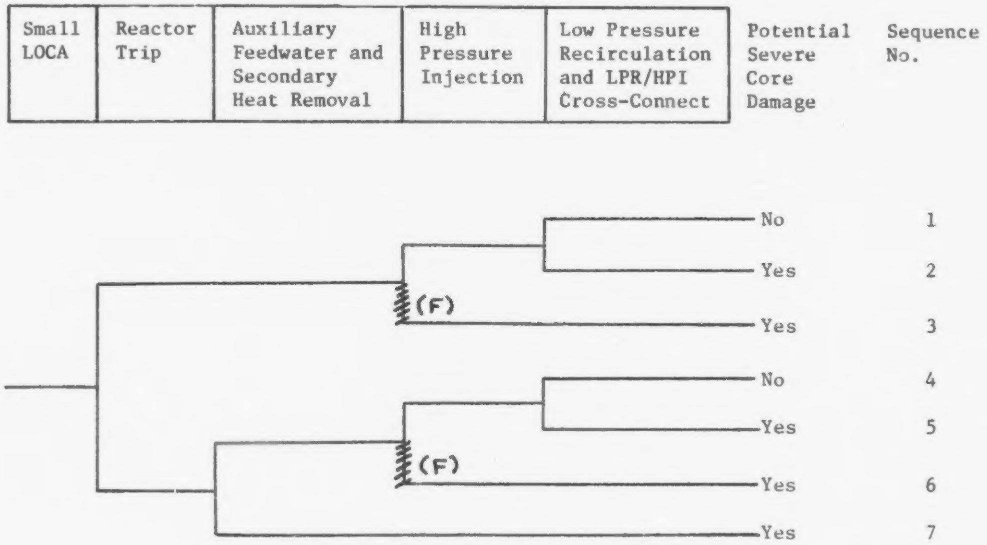
---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

AVAILABLE


Reactor in Cold Shutdown
No. 12 Safety Injection Pump Fails to Deliver Flow Due to Loose Impeller Lock Nuts Resulting in Worn Spacer Sleeves and a Bent Impeller Shaft
Impeller Lock Nuts Loose on No. 11 Safety Injection Pump
Potential Severe Core Damage
No — reactor in cold shutdown plus no LOCA
No
No

NSIC 154286 — Actual Occurrence for Safety Injection Pump Fails at Salem 1


Small LOCA
Reactor Trip
Auxiliary Feedwater and Secondary Heat Removal
High Pressure Injection
Low Pressure Recirculation and LPR/HPI Cross-Connect
Potential Severe Core Damage
Sequence No.
(F)
(F)
No 1
Yes 2
Yes 3
No 4
Yes 5
Yes 6
Yes 7

NSIC 154286 — Sequence of Interest for Safety Injection Pump Fails at Salem 1

715

[redacted]

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 154286

DATE OF LER: December 20, 1979

DATE OF EVENT: November 28, 1979

SYSTEM INVOLVED: High Pressure Injection

COMPONENT INVOLVED: High Pressure Injection Pumps

CAUSE: Loose Impeller Lock Nuts

SEQUENCE OF INTEREST: Small LOCA

ACTUAL OCCURRENCE: Failure of one pump to deliver flow and loose lock nuts on the other pump while in cold shutdown.

REACTOR NAME: Salem 1

DOCKET NUMBER: 50-272

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 1090 MWe

REACTOR AGE: 3.0 yr

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: PSE&G

OPERATORS: Public Service Electric & Gas Co.

LOCATION: 20 miles south of Wilmington, DE

DURATION: 360(a) hours

PLANT OPERATING CONDITION: cold shutdown

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________

DISCOVERY METHOD: pump breaker tripped after pump was started.

COMMENT: High Pressure Injection Pump No. 22 for Salem 2 was also found to have loose impeller locknuts. Pump 21 for Salem 2 was found to have its lock nuts secured with setscrews. The pump vendor recommended against this and the pump rotating elements were returned to the pump vendor for rework.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 154639

Date: January 5, 1980

Title: Both Diesel Generators Inoperable at Davis Besse 1

The failure sequence was:

1. With the reactor shutdown, the diesel generator 1-1 monthly test was in progress.
2. Diesel generator 1-1 experienced load swings as its output power was increased. The diesel was removed from service. The cause of the load swinging was not determined at the time the report was written.
3. During the investigation of the DG 1-1 load swing problem, an electrician racked out the DG 1-2 output breaker in lieu of the DG 1-1 output breaker, resulting in unavailability of both diesel generators for approximately 20 minutes.

Corrective action:

Oil was added to the DG 1-1 governor and the monthly test was completed successfully. Additional trouble shooting of the load swing problem was scheduled. Additional training concerning safety tagging was scheduled.

Design purpose of failed system or component:

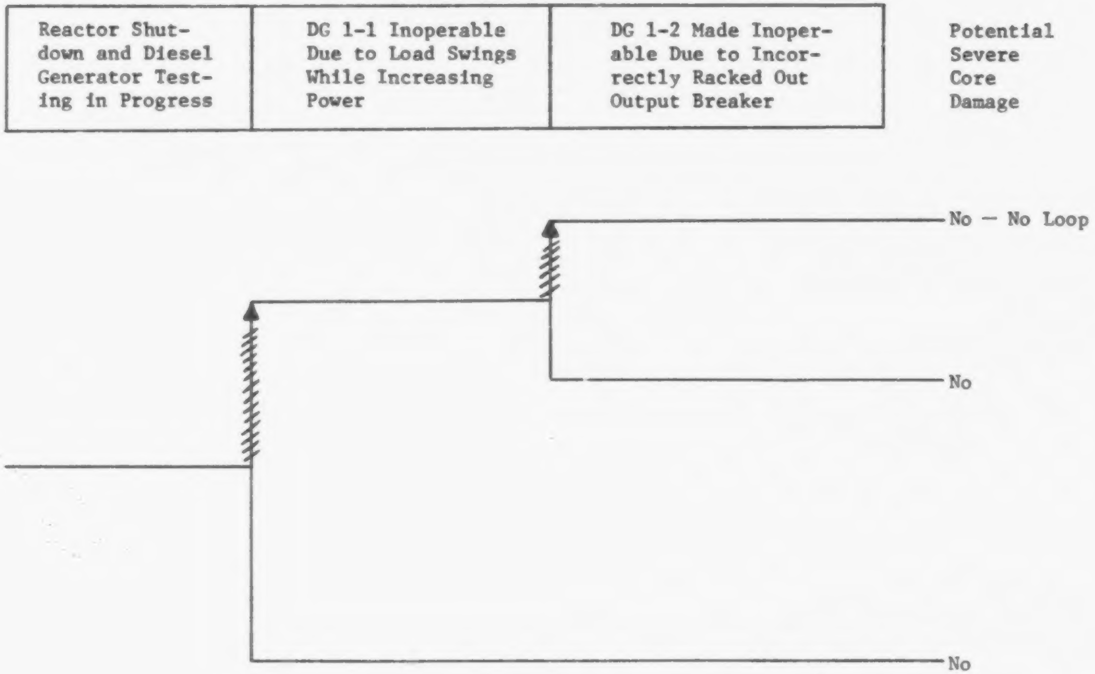
The diesel generators provide electric power for safety-related equipment when the unit generator and offsite power sources are unavailable.

Unavailability of system per WASH 1400:* Emergency power: 1×10^{-2}/D

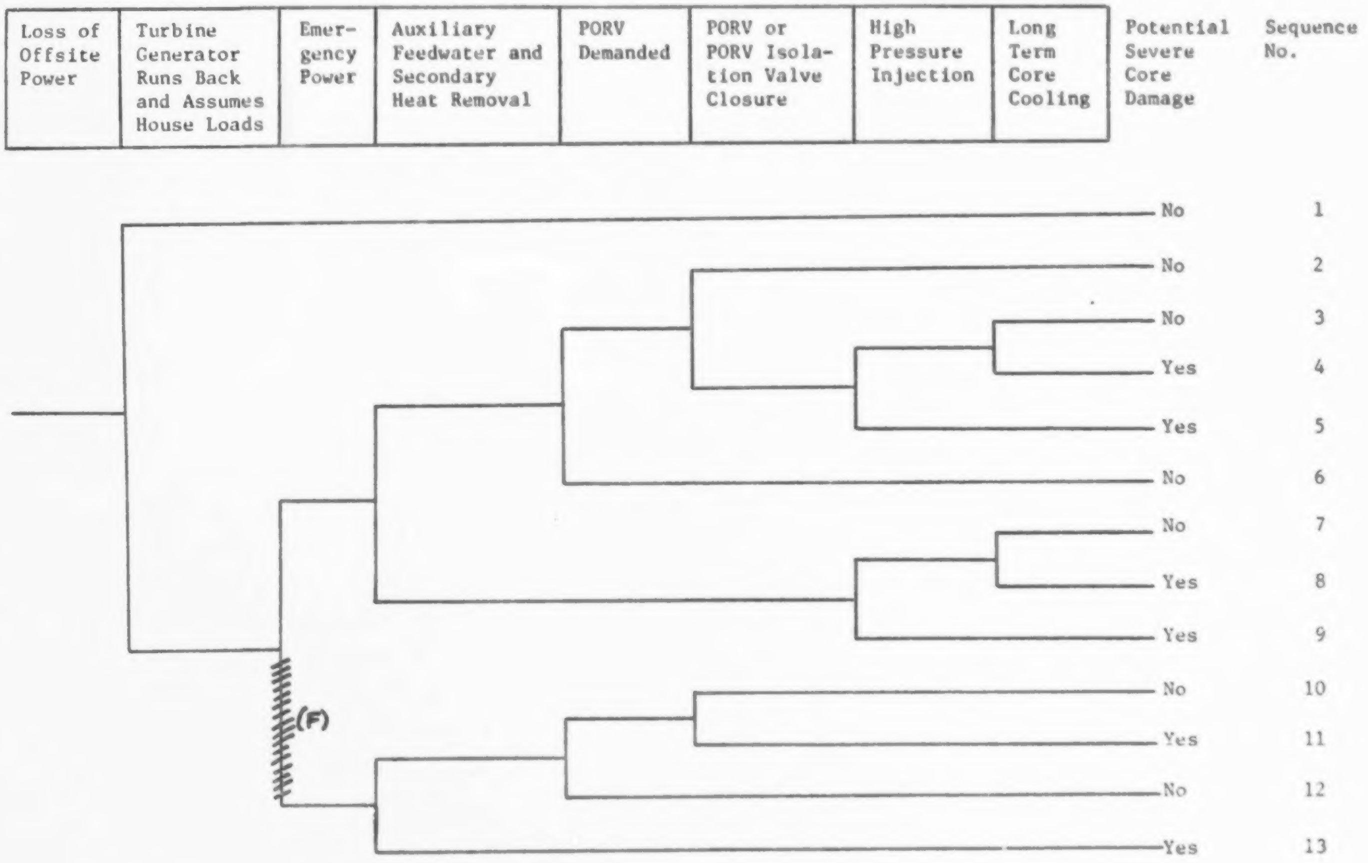
Unavailability of component per WASH 1400:* Diesel generators: 3×10^{-2}/D

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.


Reactor Shutdown and Diesel Generator Testing in Progress
DG 1-1 Inoperable Due to Load Swings While Increasing Power
DG 1-2 Made Inoperable Due to Incorrectly Racked Out Output Breaker
Potential Severe Core Damage
No — No Loop
No
No

NSIC 154639 — Actual Occurrence for Both Diesel Generators Inoperable at Davis Besse 1


Loss of Offsite Power
Turbine Generator Runs Back and Assumes House Loads
Emergency Power
Auxiliary Feedwater and Secondary Heat Removal
PORV Demanded
PORV or PORV Isolation Valve Closure
High Pressure Injection
Long Term Core Cooling
Potential Severe Core Damage
Sequence No.
No 1
No 2
No 3
Yes 4
Yes 5
No 6
No 7
Yes 8
Yes 9
No 10
Yes 11
No 12
Yes 13
(F)

NSIC 154639 — Sequence of Interest for Both Diesel Generators Unavailable at Davis Besse 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 154639

DATE OF LER: January 5, 1980

DATE OF EVENT: December 9, 1979

SYSTEM INVOLVED: Electric Power

COMPONENT INVOLVED: Diesel generators

CAUSE: Load swings on one diesel generator, uncorrectly racked out breaker on other diesel generator

SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE: Unavailability of two diesel generators with reactor shutdown

REACTOR NAME: Davis Besse 1

DOCKET NUMBER: 346

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 906 MWe

REACTOR AGE: 2.3 yr

VENDOR: B&W

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Toledo Edison

LOCATION: 21 miles East of Toledo, Ohio

DURATION: .33 hours

PLANT OPERATING CONDITION: 0% power

SAFETY FEATURE TYPE OF FAILURE: [✓ (a)] inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________

DISCOVERY METHOD: During testing

COMMENT:

NUREG/CR-2497
Volume 2
ORNL/NSIC-182/V2
Dist. Category AN, RG, XA, 1S

Internal Distribution

1. J. R. Buchanon
2. W. R. Casto
3. J. C. Cleveland
4. W. B. Cottrell
5. G. F. Flanagan
6. R. B. Gallaher
7. J. D. Harris
8. A. L. Lotts
9. G. T. Mays
10. S. H. McConathy
11. J. W. Minarick (consultant)
12. D. L. Moses
13. D. L. Phung
14. W. P. Poore
15. T. W. Robinson, Jr.
16. H. E. Trammell
17. R. Uppuluri
18. Central Research Library
19. Document Reference Section
20-21. Laboratory Records Department
22. Laboratory Records, RC
23-50. Nuclear Safety Information Center

External Distribution

51. J. Fisher, Nuclear Safety Analysis Center, Electric Power Research Institute, P.O. Box 10412, Palo Alto, CA 94303
52. C. A. Kukielka, Science Applications, Inc., 800 Oak Ridge Turnpike, Oak Ridge, TN 37830
53. A. F. McBride, Science Applications, Inc., 800 Oak Ridge Turnpike, Oak Ridge, TN 37830
54. J. R. Penland, Science Applications, Inc., 800 Oak Ridge Turnpike, Oak Ridge, TN 37830
55. E. L. Zebroski, Vice President, Institute of Nuclear Power Operations, 1820 Water Place, Atlanta, GA 30339
56. Office of Assistant Manager for Energy Research and Development, Department of Energy, Oak Ridge Operations Office, Oak Ridge, TN 37830
57-81. F. M. Manning, Division of Risk Analysis, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555
82-83. DOE Technical Information Center, P.O. Box 62, Oak Ridge, TN 37830
84-408. Given distribution as shown in category AN, RG, XA (10-NTIS).

☆U.S. GOVERNMENT PRINTING OFFICE: 1982 546-067/58

721

NRC FORM 335
(11 81)

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER *(Assigned by DDC)*
NUREG/CR-2497, Vol. 2
ORNL/NSIC-182/V2

4. TITLE AND SUBTITLE *(Add Volume No., if appropriate)*
Precursors to Potential Severe Core Damage Accident:
1969 1979 A Status Report Vol.2 - Appendix B

2. *(Leave blank)*

3. RECIPIENT'S ACCESSION NO.

7. AUTHOR(S)
J.W. Minarick C.A. Kukielka

5. DATE REPORT COMPLETED
MONTH April YEAR 1982

9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*
Science Applications, Inc.

Oak Ridge National Laboratory
Oak Ridge, TN 37830

DATE REPORT ISSUED
MONTH June YEAR 1982

6. *(Leave blank)*

8. *(Leave blank)*

12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*
Division of Risk Analysis
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

10. PROJECT TASK/WORK UNIT NO.

11. FIN NO.
B1583

13. TYPE OF REPORT
Technical

PERIOD COVERED *(Inclusive dates)*
1969-1979

15. SUPPLEMENTARY NOTES

14. *(Leave blank)*

16. ABSTRACT *(200 words or less)*

Descriptions of 169 operational events reported as Licensee Event Reports, which occurred at commercial light-water reactors during 1969—1979 and which are considered to be precursors to potential severe core damage, are presented, along with associated event trees and categorizations and subsequent analyses. The report summarizes work in (1) the development of methods used to screen ~19,400 LER abstracts for potential precursors, (2) the initial screening of those abstracts to determine which should be reviewed in detail, (3) the detailed review of those selected LERs that yielded the 169 events, (4) the categorization of the 169 events, (5) the calculation of function failure estimates based on precursor data, (6) the use of probability of severe core damage estimates to rank precursor events and estimate the frequency of severe core damage, (7) the identification of 52 events considered significant, (8) trends analyses of those significant events, and (9) the identification of the other events of interest that occurred within 1 month of significant events.

17b IDENTIFIERS OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT
Unlimited

19. SECURITY CLASS *(This report)*
Unclassified

20. SECURITY CLASS *(This page)*
Unclassified

21 NO OF PAGES

22 PRICE
S

NRC FORM 335 (11 81)

722

BEST DOCUMENT AVAILABLE